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NUREG-1137
Supplement No. 2

Safety Evaluation Report

related to the operation of Vogtle Electric Generating Plant, Units 1 and 2

Docket Nos. 50-424 and 50-425

Georgia Power Company, et al.

ORIGINAL

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

May 1986

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ABSTRACT

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and City of Dalton, Georgia, for a license to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). Supplement 1 to NUREG-1137 was issued by the staff in October 1985. The facility is located in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River.

This second supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued. This supplement also discusses some new open and confirmatory items.

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1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

In June 1985, the Nuclear Regulatory Commission staff (NRC or staff) issued a Safety Evaluation Report (SER), NUREG-1137, on the application of the Georgia Power Company (hereinafter referred to as the applicant) for licenses to operate the Vogtle Electric Generating Plant, Units 1 and 2. Supplement 1 to NUREG-1137 was issued in October 1985. This document is the second supplement to that SER (SSER 2). This supplement also provides the staff evaluation of open and confirmatory items that have been resolved and discusses new open and confirmatory items that have arisen since SSER 1 was issued.

Each of the sections and appendices of this supplement is designated the same as the related portion of the SER. Each section is supplementary to and not in lieu of the discussion in the SER unless otherwise noted. Appendix A is a continuation of the chronology of this safety review, and Appendix B lists reference materials cited in this document.* Appendix D lists acronyms and initialisms used in this supplement, Appendix E lists the principal staff contributors, and Appendix K is a continuation of errata to the SER and its first supplement. Appendices C, F, G, H, I, J, and L have not been changed by this supplement; however, one new appendix has been added. Appendix M is a Technical Evaluation Report discussing the applicant's compliance to Regulatory Guide (RG) 1.97.

In addition to updating the status of unresolved items as identified in SSER 1, this supplement

(1) discusses a Technical Specification requirement for the minimum level of the nuclear service cooling water tower basins (see Section 2.4.14)

(2) documents the staff's finding regarding Technical Specification requirements for local surveillance of each electrical tunnel (see Section 7.3.3.11)

(3) documents the staff's conclusion regarding a non-fire-rated hatch cover (see Section 9.5.1.4)

(4) documents the staff's finding regarding the applicant's request not to install a sprinkler system inside containment (see Section 9.5.1.6)

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*Availability of all material cited is described on the inside front cover of this report.

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1.7 Open Items

The SER identified 14 items in the staff review that had not been resolved with the applicant at the time that report was issued. SSER 1 fully resolved four items and partially resolved one item. This supplement fully resolves open items 4 and 6, partially resolves open items 5, 7a, and 13, expands open item 7a and identifies five new open items (7c and 15 through 18).

The complete list of open items is reproduced in updated Table 1.4; the current status of each item is given. For those items addressed in this supplement, the relevant section is noted. These issues must be sufficiently resolved before Vogtle Unit 1 fuel loading and will be reported in a future SER supplement.

1.8 Confirmatory Items

The SER identified 50 items that required confirmatory information and hence were not fully resolved at the time that report was issued. SSER 1 fully resolved nine items. This supplement fully resolves confirmatory items 2, 3, 5, 16, 17, 21, 31, 34, 37, 38, 44, and 50. This supplement also adds confirmatory item 51. The complete list of confirmatory items and their status is provided in updated Table 1.5. If the confirmatory item is discussed in this supplement, the section in which it is discussed is identified. The remaining confirmatory items must be resolved before Vogtle Unit 1 fuel loading and will be addressed in a future SER supplement.

1.9 License Condition Items

In the SER, the staff identified 11 license conditions. These issues will be cited in the operating license or Technical Specifications to ensure that NRC requirements are met during plant operation. SSER 1 added license condition 12. This supplement adds a license condition on fire protection. Table 1.6 is updated in this report to reflect this addition.

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Table 1.4 Listing of open items (revised from SSER 1)

Item	Status	SER Section*
(1) Equipment qualification	Awaiting site audit	
(a) Seismic equipment qualification		
(b) Environmental equipment qualification		
(c) Pump and valve operability assurance		
(i) Aging and sequence of environmental conditions in maintenance program		
(ii) Pumps affected by static shaft analysis		
(iii) Onsite audit		
(2) Preservice inspection program	Partially resolved (SSER 1), awaiting information, under staff review	
(3) Containment sump	Resolved (SSER 1)	
(4) Toxic gas evaluation of chemicals	Resolved (SSER 2)	6.4
(5) Generic Letter 83-28	Partially resolved (SSER 2), awaiting information, under staff review	15.8
(6) Emergency response capability--RG 1.97, Rev. 2	Resolved (SSER 2)	7.5.2.1
(7) Fire protection items		
(a) Fire doors and dampers	Partially resolved and expanded (SSER 2)	9.5.1.4
(b) Power supplies for ventilation	Awaiting information	
(c) Sprinkler system flushing deviation	Added (SSER 2)	9.5.1.5

*Section of this supplement where item is discussed.

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Table 1.4 (Continued)

Item	Status	SER Section*
(8) Safe and alternate shutdown capability	Awaiting information	
(9) Training of emergency diesel generator personnel	Resolved (SSER 1)	
(10) Diesel fuel oil storage tank cathodic protection	Resolved (SSER 1)	
(11) Licensee qualifications for operation	Awaiting information and corporate visit	
(12) Retesting of simulator response (NUREG-0737, Item I.A.2.1)	Resolved (SSER 1)	
(13) Emergency preparedness	Partially resolved (SSER 2)	13.3
(14) Human factors engineering items	Awaiting information	
(a) Detailed control room design review		
(b) Safety parameter display system		
(15) Arbitrary intermediate pipe break criteria	Opened (SSER 2)	3.6.2
(16) Spent fuel pool rack design	Opened (SSER 2)	9.1.2
(17) Training program	Opened (SSER 2)	13.2
(a) Mitigating core damage (NUREG-0737, Item I.A.2.1)		13.2.1.2
(b) Instructor qualification and requalification		13.2.1.2
(c) Licensed operator training		13.2.1.2
(d) Nonlicensed personnel training		13.2.2
(e) Records of plant personnel training		13.2.2
(18) Compliance with RG 1.94	Opened (SSER 2)	17.3

*Section of this supplement where item is discussed.

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Table 1.5 Listing of confirmatory items (revised from SSER 1)

Item		Status	SER Section*
(1)	Correlation and analysis of data from old and new meteorological towers	Awaiting information	2.3.3
(2)	Upgrade of operational meteorological measurements program	Resolved (SSER 2)	2.3.3
(3)	Atmospheric dispersion model for dose assessments	Resolved (SSER 2)	2.3.3
(4)	NSCW cooling tower seepage analysis	Awaiting information	
(5)	Details of groundwater monitoring program	Resolved (SSER 2)	2.4.12.7
(6)	Verification of FSAR commitments on compaction of Category 1 backfill	Resolved (SSER 1)	
	(a) Audit of compaction control records		
	(b) Submittal and evaluation of supplemental test results		
(7)	Submittal and evaluation of settlement records and settlement monitoring program	Awaiting information	
(8)	Foundation competency of clay marl stratum	Resolved (SSER 1)	
(9)	Steamline break analysis outside of containment	Awaiting information	
(10)	Final pipewhip and jet impingement evaluation for high-energy piping	Awaiting information	
(11)	Design documents review	Resolved (SSER 1)	
(12)	Compliance with NUREG-0737, Item II.D.1	Under staff review	
(13)	Program submittal for inservice testing of pumps and valves	Awaiting information	

*Section of this supplement where item is discussed.

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Table 1.5 (Continued)

Item	Status	SER Section*
(14) Pump and valve operability assurance	Awaiting site audit	
(a) Compliance with RG 1.148		
(b) Methods and standards for qualification		
(c) Qualification of pump and motor		
(d) Generic testing criteria for qualifying check valves		
(e) Administrative control of component qualification		
(f) Dependability of containment isolation (purge valves)		
(g) Long-term operability of deep draft pumps (IE Bulletin 79-15)		
(15) Compliance with NUREG-0737, Item II.F.2	Under staff review	
(16) Conformance to 10 CFR 50, Appendix G, criteria (PORV setpoint curve)	Resolved (SSER 2)	5.2.2.2
(17) Discrepancy between WCAP-10529 and FSAR	Resolved (SSER 2)	5.2.2.2
(18) Examination of steam generator tubes	Awaiting information	
(19) Natural circulation boration and cooldown tests	Under staff review	
(20) Target Rock valves in RVHVS	Deleted as errata (SSER 1)	
(21) Containment responses following an MSLB	Resolved (SSER 2)	6.2.1.1.1

*Section of this supplement where item is discussed.

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Table 1.5 (Continued)

Item	Status	SER Section*
(22) Operator action in event of a small-break LOCA	Awaiting information	
(23) Volumetric examination of engineered safety features systems	Awaiting information	
(24) Test of engineered safeguards P-4 interlock	Awaiting information	
(25) IE Bulletin 80-06 concerns	Awaiting information	
(26) Override of isolation signals	Awaiting information	
(27) Bypass and inoperable status panel	Awaiting information	
(28) Compliance with NUREG-0737, Item II.K.3.1	Awaiting information	
(29) Capacity of each reserve auxiliary transformer to start and run the loads of both Class 1E trains	Awaiting information	
(30) Verification test results for the adequacy of plant electric distribution system voltages	Awaiting information	
(31) Coordination and testing of circuit breakers located in the primary circuit of regulated transformers used as isolation devices	Resolved (SSER 2)	8.4.5
(32) Plant-specific procedure for estimating core damage (NUREG-0737, Item II.B.3)	Resolved (SSER 1)	
(33) Demonstrate effective communications	Awaiting information	
(34) Offsite communications	Resolved (SSER 2)	9.5.2.2
(35) Procedures for load following test and/or maintenance	Awaiting information	

*Section of this supplement where item is discussed.

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Table 1.5 (Continued)

Item	Status	SER Section*
(36) Incorporation of generic and plant-specific recommendations for TDI diesel generators	Awaiting information	
(37) Procedures for ordering fuel after 5 days	Resolved (SSER 2)	9.5.4.2
(38) Procedures for general housekeeping and maintenance	Resolved (SSER 2)	9.5.8
(39) Process control program	Awaiting information	
(40) Volume reduction system (VRS)	Under staff review, awaiting information	
(a) VRS topical report		
(b) Potential accidents involving the VRS		
(c) VRS inputs		
(d) VRS filter testing		
(41) Compliance with NUREG-0737, Item II.K.3.31	Awaiting information	
(42) Compliance with NUREG-0737, Item II.F.1	Awaiting information	
(43) Compliance with NUREG-0737, Item III.D.1.1	Awaiting information	
(44) Procedures generation package	Resolved (SSER 2)	13.5.2
(45) Program to minimize post-LOCA leakage from ESF system outside containment	Deleted as errata (SSER 1)	
(46) Analysis of dropped control rod event for DNB limits	Resolved (SSER 1)	
(47) Inadvertent boron dilution during modes 3, 4, and 5	Resolved (SSER 1)	

*Section of this supplement where item is discussed.

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Table 1.5 (Continued)

Item	Status	SER Section*
(48) Operator action in event of an SGTR	Under staff review	
(49) Radiological consequences of an SGTR	Under staff review	
(50) Program to minimize ECCS equipment leakage	Resolved (SSER 2)	15.6.5.1
(51) Generic Letter 85-12	Opened (SSER 2)	15.1.5.1

*Section of this supplement where item is discussed.

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Table 1.6 Listing of license conditions (revised from SSER 1)

Item	SER Section*
(1) Long-term groundwater and settlement monitoring requirements	
(2) Inservice testing of pumps and valves	
(3) Final baseline report for the loose parts monitoring system	
(4) Technical Specification for maximum permissible temperature mismatch	
(5) Inservice inspection program	
(6) Operability requirements for vent system in Technical Specifications	
(7) Exemption for 10 CFR 50, Appendix J, Paragraph III.D.2(b)(ii)	
(8) Exemption from 10 CFR 70.24	
(9) Operating experience on shift	
(10) Implementation and maintenance of physical security plan	
(11) Technical Specification to require four valves to be closed during refueling	
(12) Reactor vessel level instrumentation system implementation report	
(13) Fire protection	9.5.1.7

*Section of this supplement where item is discussed.

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2 SITE CHARACTERISTICS

2.3 Meteorology

2.3.3 Onsite Meteorological Measurements Program

In the SER, the staff identified three confirmatory items related to the onsite meteorological measurements program. These items were

- correlation of data collected on old and new meteorological towers (confirmatory item 1)

- appropriateness and quality of meteorological data available for use in emergency response as part of the upgraded onsite meteorological measurements program (confirmatory item 2)

- incorporation of current meteorological data into a real-time atmospheric dispersion model for dose assessments in the context of emergency response (confirmatory item 3)

By letter dated November 12, 1985, the applicant provided data from the first full year of operation of the new meteorological tower (February 1, 1984-January 31, 1985) with a correlation of these data with previous data provided in the Final Safety Analysis Report (FSAR) in response to confirmatory item 1. To facilitate comparison between data collected on the new tower with previous data, the applicant utilized identical levels of measurement for the following parameters: windspeed and direction (10 m and 45 m), dry bulb and dew point temperature (10 m), and precipitation (near the ground). However, the interval for measurement of vertical temperature difference was originally between 45 m and 10 m at the old tower but was changed to between 60 m and 10 m at the new tower. The two towers have essentially the same siting conditions, and the data collected on the new tower for the period February 1, 1984-January 31, 1985 compare reasonably well (i.e., within expected year-to-year variability) with data collected at the same levels for earlier periods. The most significant differences among the data sets were for atmospheric stability classifications based on vertical temperature difference. Occurrences of extremely unstable (Pasquill Type "A") conditions dropped sharply from about 18% per year averaged during earlier periods to less than 8% for the most recent 1-year period. This decrease in extremely unstable conditions was generally compensated by corresponding increases in moderately unstable (Pasquill Type "B") and slightly unstable (Pasquill Type "C") conditions so that total occurrences of all unstable conditions (Pasquill Types "A," "B," and "C") for the most recent period were only slightly below the 23% per year averaged for the earlier data sets. The other notable difference in classification of atmospheric stability was the shift away from the predominance of slightly stable (Pasquill Type "E") conditions observed on the old tower (about 32% per year on the average) to neutral (Pasquill Type "D") conditions observed on the new tower (30.5% for the 1-year period). These differences may be attributable to year-to-year variability together with the change in interval for the measurement of vertical temperature

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difference, as the applicant maintains, and/or to some heretofore undetected instrumentation problems either with the previous measurement program or with the new program. Reviews of the new measurements program and the maintenance and calibration procedures have not identified any obvious errors, deficiencies, or inconsistencies that would result in problems with the measurement of vertical temperature difference. Differences attributable primarily to the change in measurement interval should be reflected in occurrences of moderately stable (Pasquill Type "F") and extremely stable (Pasquill Type "G") conditions as well as in occurrences of unstable and neutral conditions. However, no major differences are indicated between data on the old tower and data on the new tower for moderately and extremely stable conditions. The applicant must provide data summaries from the second year of operation of the new meteorological tower (February 1, 1985-January 31, 1986) to demonstrate the relative year-to-year consistency of stability classification based on the measurement of vertical temperature difference on the new meteorological tower. This comparison remains a confirmatory item.

The difference between the measurement of precipitation at the site and at Augusta was discussed in the SER. The annual average total precipitation measured at the site for the composite period December 1972-November 1973 and April 1977-March 1979 is 26.6 in. compared with the annual average of 40.7 in. at Augusta. In the November 12, 1985, letter, the applicant acknowledged that "in future years with better instrumentation and maintenance, the precipitation totals at Vogtle should approach those at Augusta." The staff will continue to monitor the quality of meteorological data collected at Vogtle by reviewing information contained in the Semi-Annual Effluent Reports and routine inspection activities.

During the Emergency Preparedness Implementation Appraisal (EPIA) conducted in March 1986, the appropriateness and quality of meteorological data available for use in emergency response (confirmatory item 2) were reviewed as part of the upgraded meteorological measurements program. The upgraded meteorological measurements program (which includes the new tower, the old tower used as a backup, and improved data transmission and reduction techniques) was found to be adequate for use in emergency response situations because it meets the intent of the guidance contained in Regulatory Guide (RG) 1.101. On this basis, the staff concludes that confirmatory item 2 is adequately resolved. The arrangements for transmission of the data and subsequent display of the information were incomplete at the time of the EPIA. These items, plus other items concerning improved procedures related to atmospheric modeling, are identified in Section 4.2.1.4 of the inspection report documenting the EPIA (Inspection Report 50-424/86-12 and 50-425/86-18 dated May 9, 1986). Issues related to meteorological data used in emergency response identified during the EPIA will be followed as inspection report items.

Also reviewed during the EPIA was the incorporation of current meteorological data into a real-time atmospheric dispersion model for dose assessments in the context of emergency response (confirmatory item 3). The applicant will have the capability to use current meteorological data in a real-time atmospheric dispersion model once the arrangements for transmission of the data from the onsite tower are completed and appropriately displayed. The staff concludes that this capability is adequate, therefore resolving confirmatory item 3. Issues related to atmospheric dispersion modeling used in emergency response will be followed as inspection report items.

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2.4 Hydrologic Engineering

2.4.12 Groundwater

2.4.12.7 Groundwater Monitoring Program

In the SER, the staff discussed the groundwater monitoring program, which was designed to monitor groundwater levels in the water table aquifer, plant backfill, Blue Bluff marl, and confined aquifer.

As stated in the SER, the number and locations of monitoring wells proposed by the applicant for the water table aquifer and plant backfill were acceptable to the staff. Figure 2.24 in the SER, however, did not show the locations of four new wells proposed by the applicant for the plant backfill. SER Figure 2.24 has been revised and is included as Figure 2.1 in this supplement to show the locations of all the wells currently being used in the groundwater monitoring program. In addition to monitoring groundwater levels in 13 wells in the water table aquifer and in 4 wells in the plant backfill, the applicant committed to install continuous water level recorders in two of these wells: one in the plant backfill and one in the water table aquifer. Readings obtained from these recording wells were to be compared with recorded rainfall amounts to determine whether there was any correlation between rainfall and groundwater level rise. The wells not having recorders were to be read weekly.

At the staff's request, the applicant by letter dated May 21, 1985, committed to drill six additional bore holes in the Blue Bluff marl. These holes were to be arranged in two clusters of three holes. Each cluster would be located at opposite corners of the power block. The location of these clusters is shown on Figure 2.1 of this supplement.

In situ permeability tests were to be conducted on the cored holes, and water levels were to be read weekly. In addition, the applicant also committed to submit a report evaluating the data obtained from the monitoring program after obtaining 6 months of data.

The staff also required that the reasonableness of each groundwater level reading be ascertained at the time of the reading by comparing it with previous readings in the same well or with present readings in adjacent wells. Where a reading was determined to be invalid, another reading was to be obtained within 2 days.

As stated in the SER, the staff found the applicant's groundwater monitoring program commitments satisfactory. Final details of the program, however, were a confirmatory item. By letter dated May 31, 1985, the applicant provided the final details of the program. These details included a discussion of where and how new wells would be drilled, the drilling equipment that would be used, how core samples would be preserved, how the in situ permeability tests would be conducted, and at what elevations within the Blue Bluff marl groundwater levels would be monitored.

The applicant discussed the procedures to be used to drill the two well clusters in the marl. Each cluster was to consist of three observation wells open to discrete intervals in the marl.

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Drilling of the new wells would proceed in two steps. First a 10-in.-diameter hole would be drilled using a tricone bit from the ground surface to a depth of 10 ft below the upper contact of the marl. At this point, the hole would be cased and the annulus between the soil and the casing would be sealed with cement grout and allowed to set for 24 hours before the drilling operation would continue. In the second step, the hole would be advanced by diamond core drilling methods using clear water as the circulating fluid. Each well would then be cased in the marl, and a porous stone piezometer would be installed to measure hydrostatic pore pressure within the marl.

The first hole in each cluster was to be drilled to within 5 to 10 ft of the base of the marl. This hole was to be continuously cored and in situ permeability tests conducted as the core hole was advanced. All core samples in good condition would be preserved in boxes in conformance with American Society for Testing and Materials (ASTM) D2113. Each box would be labeled with the bore hole number, box number, and depth interval at which the core sample was taken. Each sample would be photographed and logged in detail. Once the drilling, coring, and testing operations were completed, this primary hole would be used as the deep well to monitor the lower portion of the marl. On the basis of data collected during drilling of the first well, the location of the monitoring intervals of the other two wells in each cluster would be determined. These two wells would monitor the upper and middle intervals of the marl. The wells in each cluster would be spaced a minimum of 10 to 15 ft apart. The program was initiated during the summer of 1985.

On June 19, 1985, staff members met with the applicant at the plant site to observe the actual core drilling operation and the in situ permeability testing of the marl. The staff found the details of the groundwater monitoring program acceptable except for the location of the recording well in the water table aquifer. In a letter dated October 10, 1985, the staff advised the applicant that the selection of well 808 as the water table aquifer recording well did not appear to be appropriate because this well, which is located north of the plant, would be monitoring groundwater levels about 5 to 10 ft lower than wells on the south side of the power block area. The staff suggested that either well 803A or 804 appeared to be more appropriately situated with respect to an SER criterion that the recording wells be located "...relative to...previously measured high groundwater levels...." The staff required in its October 10, 1985, letter that the applicant discuss how the dynamic response of recording well 808 could be combined with the weekly measurements from other wells and extrapolated to the design-basis groundwater level.

In a letter dated November 15, 1985, the applicant responded to the staff's concern. In this letter, the applicant stated that because wells 803A and 804 are located upgradient from the power block area, groundwater levels in these wells would be expected to always be higher than the groundwater level at critical plant structures. A better indicator of groundwater levels at critical structures would be well LT-13, which is also a continuous recording well and is located in the power block area. The applicant stated that the well 808 record would provide a second correlation with rainfall for comparison with the well LT-13 record.

The staff reviewed the applicant's November 15, 1985, letter and concluded that a continuous recorder at well 808 is acceptable if the applicant is able to

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establish a relationship between rainfall and groundwater level rise at safety-related structures using data recorded at wells 808 and LT-13. However, if this relationship cannot be determined, the staff may require that the recording wells be relocated.

On August 23, 1985, the applicant submitted a report detailing the results of the field and laboratory studies that had been initiated in the summer of 1985. This report discussed the in situ packer permeability tests that had been conducted in the six new bore holes in the marl. These new wells are shown on Figure 2.1 as wells 900, 901, 902, 903, 904B, and 905. Packer tests were conducted at 10-ft intervals throughout the entire depth of the marl.

One of the staff's concerns with in situ permeability tests conducted during site exploration (1971-1973) was that some of the holes had been drilled using bentonite, which is a colloidal clay, as the circulating fluid. The staff's concern was that the bentonite could have plugged the more permeable zones in the marl, thus resulting in calculated permeabilities that were lower than those that actually existed. To alleviate this concern, the applicant used only clear water when drilling the six new bore holes in the marl.

In all the packer tests conducted, the water takes were zero, indicating an apparent permeability of zero. Thus, packer tests could not be used to quantify the permeability of the marl because of its very low values. To quantify marl permeability, laboratory measurements were made on 10 marl samples obtained during core drilling. A falling head permeability test was conducted on each sample. The results of these tests showed the permeability of the marl to range from about 10^{-8} cm/sec to 10^{-6} cm/sec. These laboratory tests together with the packer field tests confirmed that the marl is essentially impermeable. The six new bore holes in the marl were screened at varying depths, and piezometers were installed so that water levels can be monitored throughout the full depth of the marl.

Included in the August 1985 report were drill logs for the six new bore holes drilled in the marl. These logs indicate that there are no voids in the marl because no rod drops were recorded. Neither did the logs show any loss of circulating water in the holes as would have been expected if pervious materials or marl fractures had been encountered which would have allowed drill water to leak out. Thus, the August report provided evidence that the marl is continuous and that there are no detectable paths for water to leak into the confined aquifer beneath the marl.

Also discussed in the August 1985 report were six new wells that were installed in the water table aquifer. The six new wells are identified on Figure 2.1 of this supplement as 808, 809, LT-12, LT-13, LT-7A, and LT-1B. Two of these wells, LT-7A and LT-1B, replaced two existing wells, LT-7 and LT-1A, which had been damaged during the backfilling operation and could not be used as monitoring wells. The damaged wells were grouted from the bottom up and were abandoned. As requested by the staff, two wells were equipped with continuous water level recorders: well LT-13 in the plant backfill and well 808 in the water table aquifer.

On the basis of a review of the final details of the applicant's groundwater monitoring program as discussed in the applicant's May 31, 1985, and August 23, 1985, submittals, the staff concludes that the applicant has adequately

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addressed the staff's concern regarding confirmatory item 5. The staff thus considers the groundwater portion of this issue resolved. The portion of this item dealing with settlement monitoring was combined with confirmatory item 7 as discussed in SER Supplement 1.

In accordance with its commitment to prepare and submit a report after obtaining 6 months of groundwater level data from its groundwater monitoring program, the applicant provided its report by letter dated February 6, 1986. This report contains the data obtained during the period July-December 1985. The results of the first 6-month monitoring period show that the amount and rate of water level rise or decline vary from place to place. The largest changes were recorded in wells 807A and 808, which are located in the water table aquifer north of the power block, and in well LT-12, which is located in the plant backfill. The applicant states that the large groundwater level rises recorded in these wells were due to temporary construction conditions. Wells 807A and 808 were in the area of a large depression that ponded water and abnormally increased the amount of recharge. Well LT-12 was located in a small, enclosed low area that also concentrated the source of recharge. The applicant expects that once the site grading and drainage plan is completed, the groundwater level rises in these wells will be much less than what was observed during the initial 6-month monitoring period. The balance of the observation wells in the water table aquifer and plant backfill showed a relatively steady rise except for wells 804 and 179, which showed a relatively constant elevation during the 6-month period.

As discussed above, the staff required that the reasonableness of groundwater level readings be checked, and if found to be invalid, another reading was to be obtained within 2 days. The applicant points out in its August report that wells 801, 806B, and 807A had readings that deviated significantly from the general trend and were probably invalid readings. However, these measurements were not checked as required by the staff. Therefore, the staff will require that these three wells continue to be monitored weekly as discussed below.

The applicant compared the groundwater level records obtained from the wells having continuous recorders with recorded rainfall values. This comparison established that there is no immediate response to individual rainstorms. However, a persistently wet period of a month or more can result in a detectable rise in groundwater levels.

Plotted hydrographs of the observed groundwater level data show that a general rise in the water table of as much as 0.5 ft in a period as short as 1 month has not occurred, although localized fluctuations of 0.5 ft or more have occurred in as short a time as 1 day. The applicant states, however, that these short-term rises may have been influenced by construction activities.

On the basis of the data obtained during the initial 6-month monitoring period, the applicant concludes that a general rise of the water table of 1 ft or more requires at least a seasonal period (3 to 6 months) of markedly wet weather. Such a trend would be readily identified by monthly monitoring instead of the weekly monitoring now required by the staff. Thus, the applicant requests that until site grading is complete, the two continuous recorders (808 and LT-13) be maintained and the other three wells in the backfill adjacent to the structures (LT-1B, LT-7A, and LT-12) be continued to be monitored weekly. The applicant further requests that the monitoring frequency of the balance of wells in the

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water table aquifer be reduced to once each month. This monitoring will coincide with monitoring of the confined aquifer observations wells. Monitoring of the piezometers in the marl will continue to be monitored weekly through March 1986. This monitoring schedule will confirm the trend and fluctuation characteristics of the pore pressure in the marl, as indicated by the first 6 months of record. Once the trend at equilibrium is established, the applicant requests that monitoring be reduced to monthly readings.

On the basis of a review of the February 6, 1986, submittal, the staff concludes that a reduction in the monitoring frequency of some of the wells is warranted. The staff agrees that the two continuous recorders (808 and LT-13) should be maintained and that weekly monitoring of the other three wells in the plant backfill be continued. The staff, however, does not agree with the applicant that the monitoring frequency of the balance of the water table aquifer wells be reduced to once a month because construction activities may have affected water level responses in certain wells. The staff will thus require that, in addition to maintaining the two continuous recorders and weekly monitoring of the three wells in the plant backfill, wells 800, 801, 806B, 807A, and 808, whose readings were affected by inadequate grading, also be monitored weekly. The staff agrees that monthly monitoring of the remainder of the wells is acceptable.

2.4.14 Technical Specifications and Emergency Operating Requirements

As stated in SER Section 2.4.11, the ultimate heat sink (UHS) water supply for Vogtle is the water stored in the nuclear service cooling water mechanical draft cooling tower basins. Makeup water to replenish losses from the UHS system is obtained from either groundwater or from the Savannah River.

Basin water levels are normally maintained at elevation 217 ft 9 in., with a low level alarm set to operate whenever the basin levels fall approximately 6 in. to a water level of 217 ft 3 in. At this lower level, the cooling tower basins store sufficient water to meet the requirements of RG 1.27, "Ultimate Heat Sink for Nuclear Power Plants." To ensure an adequate supply of water, the staff requires a technical specification to define the actions to be taken whenever the water level in the cooling tower basins falls below elevation 217 ft 3 in. The applicant has submitted draft Technical Specifications that include a provision to address this requirement. The staff will ensure the adequacy of the applicant's submittal in this area during a detailed Technical Specification review.

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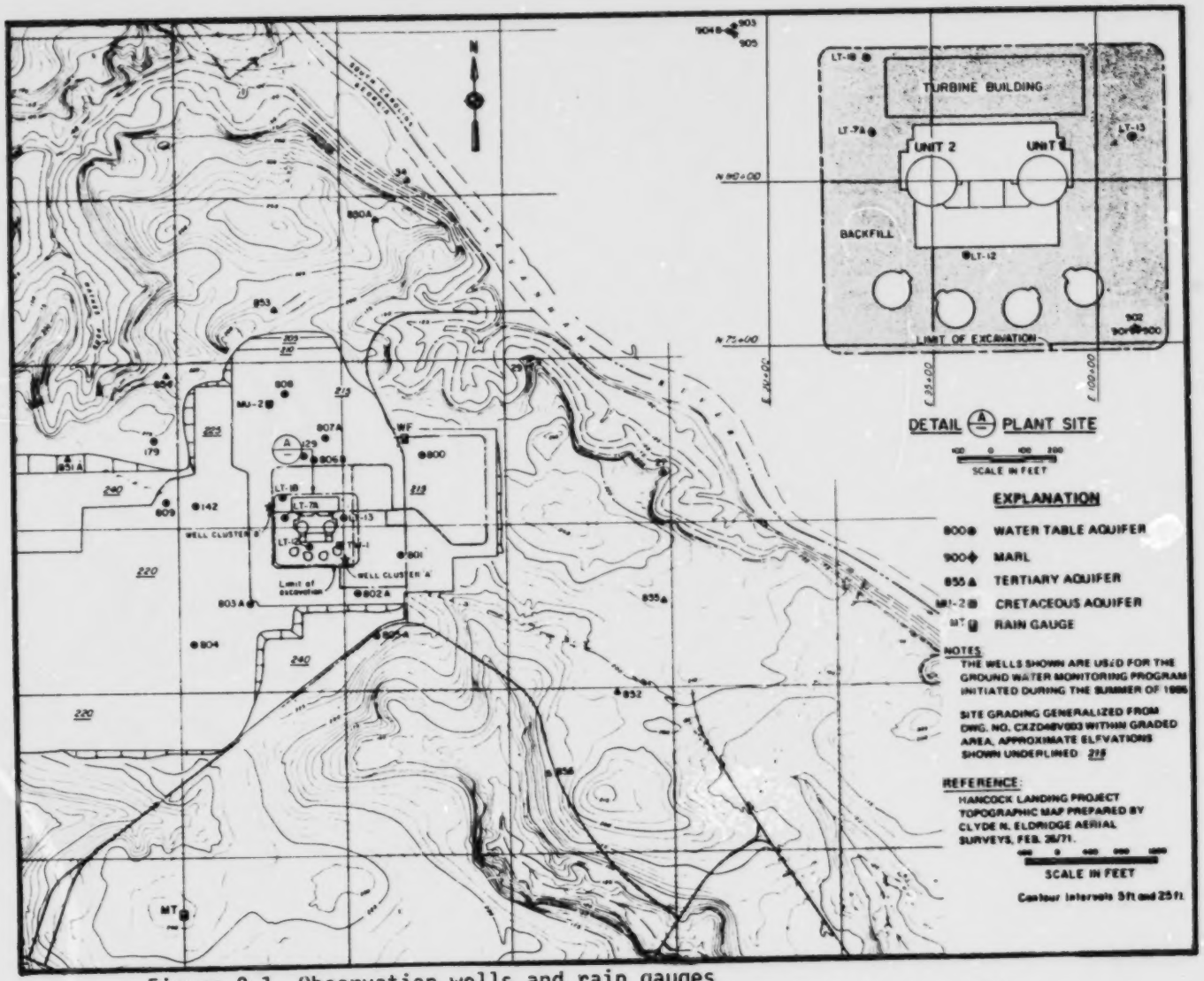
TURBINE BUILDING
UNIT 2
UNIT 1
BACKFILL
LIMIT OF EXCAVATION
DETAIL A PLANT SITE
SCALE IN FEET
EXPLANATION
800 WATER TABLE AQUIFER
900 MARL
855 TERTIARY AQUIFER
MW-2 CRETACEOUS AQUIFER
MT RAIN GAUGE
NOTES
THE WELLS SHOWN ARE USED FOR THE GROUND WATER MONITORING PROGRAM INITIATED DURING THE SUMMER OF 1985
SITE GRADING GENERALIZED FROM DWG. NO. CXZD485V003 WITHIN GRADED AREA, APPROXIMATE ELEVATIONS SHOWN UNDERLINED 215
REFERENCE:
HANCOCK LANDING PROJECT TOPOGRAPHIC MAP PREPARED BY CLYDE N. ELDRIDGE AERIAL SURVEYS, FEB. 26/71.
SCALE IN FEET
Contour intervals 5ft and 25ft
WELL CLUSTER 'A'
WELL CLUSTER 'B'
Limit of excavation
MT

Figure 2.1 Observation wells and rain gauges
Source: Bailey, J. A., GPC, letter to H. R. Denton, NRC, "SER Confirmatory Item 5: Groundwater Monitoring Program," February 6, 1986

3 DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT, AND SYSTEMS

3.6 Protection Against Dynamic Effects Associated With the Postulated Rupture of Piping

3.6.2 Determination of Rupture Locations and Dynamic Effects Associated With the Postulated Rupture of Piping

In the SER, the staff indicated that approval of the applicant's proposals (November 11, 1983, April 26, 1984, and April 24, 1985) for eliminating arbitrary intermediate pipe breaks was contained in letters dated June 28, 1984, and June 13, 1985. These approvals were based in part on the applicant's statement in its April 26, 1984, letter, that no welded attachments were within five pipe diameters of postulated break locations. Subsequently, the staff identified in IE Inspection Report 50-424/85-35 dated December 18, 1985, an unresolved item regarding lack of controls to preclude welded attachments from being placed within five pipe diameters of a postulated break location. Further, an inspection conducted at Vogtle during the week of May 5, 1986, revealed that welded attachments are located within five pipe diameters of six postulated break locations that the applicant requested to be eliminated. Because the staff's approvals were based on the applicant's statement in its April 26 letter, the applicant is required to revise its earlier proposals to contain a detailed description of how local stresses at these attachments were accounted for. Until the staff receives and reviews this information, the issue of arbitrary intermediate pipe breaks is considered an open item that must be resolved before licensing.

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5 REACTOR COOLANT SYSTEM

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.2 Overpressure Protection

5.2.2.2 Overpressure Protection During Low-Temperature Operation

As discussed in the SER, the staff reviewed the design of the system at Vogtle that prevents the reactor system pressure from exceeding the limits imposed by Appendix G to 10 CFR 50. The Vogtle units utilize the cold overpressure mitigation system (COMS) designed by Westinghouse, which is described in Topical Report WCAP-10529, "Cold Overpressure Mitigating System." The staff determined that the COMS as backed up by pressure relief valves in the two residual heat removal (RHR) suction lines would prevent the allowable temperature-pressure limits from being exceeded using assumptions of worst-case mass addition or worst-case heat input.

Although the design of the COMS at Vogtle was determined to be acceptable, the staff identified two confirmatory items. Confirmatory item 16 required that the applicant develop an acceptable plant-specific power-operated relief valve (PORV) setpoint curve to be incorporated in the Technical Specifications. By letter dated February 28, 1986, the applicant submitted its draft Technical Specifications containing the Vogtle PORV setpoint curve. The curve's range of temperatures is in conformance with Appendix G to 10 CFR 50. On this basis, the staff concludes that the Vogtle PORV setpoint curve is acceptable. Therefore, confirmatory item 16 is fully resolved.

Confirmatory item 17 required that a discrepancy between the FSAR and WCAP-10529 regarding the actual temperatures used for the worst-case heat input transient be corrected. Temperature is not a significant parameter for mass input transients; hence this information was not necessary for the mass input case. In Amendment 19 to the FSAR, the applicant indicated that the range of actual temperatures for the worst-case heat input transient is 70°F to 280°F for the reactor coolant system. The corresponding range for the steam generator for this case is 120°F to 330°F. The staff has reviewed this information and concludes that it is satisfactory to fully resolve confirmatory item 17.

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6 ENGINEERED SAFETY FEATURES

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Containment Analysis

6.2.1.1.1 Maximum Containment Pressure and Temperature Analysis

Secondary System Breaks

In the SER, the staff identified the mass and energy release data for postulated main steamline break (MSLB) accidents as a confirmatory item pending the receipt of additional information concerning the Westinghouse methodology and its applicability to Vogtle. By letter from E. P. Rahe, Westinghouse, to H. R. Denton, NRC, dated October 7, 1985, Topical Report WCAP-8822-S2 (and the nonproprietary version, WCAP-8860-S2), "Mass and Energy Release Following a Steamline Rupture," were submitted for staff review. This supplement to the WCAP report specifically addresses staff concerns regarding the ability of large, dry containments to withstand pressures and temperatures associated with superheated steam blowdown. By letter dated May 27, 1986, from C. E. Rossi, NRC, to E. P. Rahe, Westinghouse, the staff discussed its evaluation and acceptance of Supplements 1 and 2 to WCAP-8822 for referencing in licensing actions to the extent specified in the evaluation. In essence, a superheated steam blowdown does not significantly affec. pressure-temperature response of a typical large, dry containment. The staff found the Vogtle containment parameters are bounded by the containment described in the Westinghouse topical report, with the exception of delayed spray initiation. By letter from E. P. Rahe, Westinghouse, to H. R. Denton, NRC, dated December 13, 1985, Westinghouse showed that further delaying the initiation of containment spray will not adversely affect the containment response and, hence, the conclusions of WCAP-8822-S2. On the basis of the findings of the Westinghouse topical report and the demonstration of its applicability to Vogtle, the staff concludes that the mass and energy release data for the postulated MSLB accident are acceptable, and that confirmatory item 21 is satisfactorily resolved.

6.4 Control Room Habitability

The SER identified the toxic gas protection of control room operators as an open item. The open item included both the applicant's departure from RG 1.95 (the control room isolated air exchange rate is eight times higher than the RG value) and the ability of operators to smell chemicals and don breathing apparatus. In a meeting held on July 31, 1985, the applicant presented a new analysis of chlorine dispersion taking credit for the fact that chlorine is more than two times heavier than air and for the height differential between the ground and air intakes. The staff indicated to the applicant that its analysis as presented did not appropriately account for wake effects from intervening buildings or point source releases at ground level associated with neutral or unstable atmospheric diffusion conditions. The staff directed the applicant to

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perform two types of analyses. These analyses--a building wake model and a flat plain analysis--were clarified in a September 6, 1985, letter from E. G. Adensam, NRC, to D. O. Foster, GPC.

The applicant examined a number of atmospheric dispersion models to characterize the transport and diffusion of chlorine resulting from the rupture of a 1-ton cylinder stored on site at the nuclear service cooling water chlorine storage building. The applicant's various analyses were discussed in submittals dated November 26, 1985, and January 17, 1986. The rupture of such a cylinder is typically assumed to result in a near-instantaneous "puff" release followed by a longer-term "continuous" release until the cylinder is empty (generally on the order of several hours, depending on meteorological conditions). The behavior of chlorine released immediately following a cylinder rupture is difficult to simulate because of the energetics associated with the rupture itself and the rapid entrainment of air into the chlorine cloud produced. Although chlorine is heavier than air, the rapid entrainment of air into the chlorine cloud probably results in essentially neutral buoyancy within a short time period. The characterization of the dispersal of chlorine resulting from a cylinder rupture at Vogtle is further complicated by the presence of a variety of structures located around the cylinder storage area, between the cylinder storage area and the control room air intakes, and around the control room air intakes. The presence of these structures affects both the airflow pattern and atmospheric turbulence in the vicinity of the structures. The regulatory position presented in RG 1.95 was based on a relatively simple atmospheric dispersion model (described in Appendix B of RG 1.78) intended to be reasonably conservative in characterizing transport and diffusion following a cylinder rupture in the vicinity of plant structures for situations where the control room air intakes were within a relatively short distance of the release point. In the November 26, 1985, and January 17, 1986, submittals, the applicant examined several alternatives to the RG 1.95 approach, but concluded that other available models do not adequately characterize the complexities of transport and diffusion given the building configuration and transport path at Vogtle. In particular, the applicant concluded that the model used to simulate the effects of turbulent mixing in the vicinity of plant structures produced excessive peak chlorine concentrations at the locations of the control room air intakes. The staff agrees that the alternative models examined by the applicant are not appropriate to the situation at Vogtle. Therefore, the staff concludes that the approach described in RG 1.95 for characterizing atmospheric dispersion following the rupture of a chlorine cylinder is appropriate for use at Vogtle.

In using the RG 1.95 approach, the applicant assumed that the relative concentration (χ/Q), which was equalled or exceeded 5% of the time at Vogtle, was equivalent to that calculated considering moderately stable (Pasquill Type "F") stability conditions accompanied by a windspeed of 0.9 m/sec. This assumption is not fully supported by information presented in Table 3 of the evaluation that accompanied the applicant's November 26, 1985, letter nor by information presented in the FSAR. For example, information presented in Table 3 of the applicant's November 26, 1985, letter indicates that atmospheric dispersion conditions that produce χ/Q values that are equal to or that exceed those calculated (using the RG 1.95 approach) assuming Pasquill Type "F" stability with a windspeed of 0.9 m/sec occur about 8.8% of the time. An independent evaluation performed by the staff determined that the 5% percentile atmospheric dispersion condition was equivalent to that calculated considering Pasquill Type "F" stability accompanied by a windspeed of 0.7 m/sec. The applicant also assumed that

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the chlorine was uniformly mixed to a height of 18.4 m (the elevation of the air intakes) rather than the height of 15 m considered in RG 1.95. The staff agrees that the use of the plant-specific information related to the elevation of the air intakes is more appropriate in this regard. Furthermore, at the request of the staff, the applicant presented an additional analysis for Pasquill Type "F" stability accompanied by a windspeed of 0.7 m/sec in its January 17 letter. The staff accepts the atmospheric dispersion evaluation presented in the applicant's January 17, 1986, letter based on the RG 1.95 model, modified to consider uniform mixing to a height of 18.4 m, with moderately stable conditions accompanied by a windspeed of 0.7 m/sec.

The applicant also committed to perform a reanalysis of the infiltration of chlorine gas from outside air into the control building. This reanalysis was submitted to the staff by letter dated February 19, 1986. Using standard assumptions contained in RGs 1.78 and 1.95 and conservative assumptions where the uniqueness of the Vogtle heating, ventilation, and air conditioning (HVAC) system design prevented use of standard assumptions, the applicant performed an analysis that showed that chlorine concentration within the control room would not exceed the guideline 15-ppm limit as stated in RGs 1.78 and 1.95.

Because the Vogtle control building is served by several HVAC systems and because areas adjoining the control room, under certain conditions, are capable of transporting contaminated air to the control room, a plant-specific infiltration model had to be developed to describe the transport of chlorine gas into the control room area. A nodal analysis was performed in which air volumes capable of providing inleakage into the control room were grouped into seven nodes. A mass balance was then set up between these, taking into consideration mixing and dilution followed by infiltration from each node into the control room. Chlorine concentrations as a function of time were calculated for each node and the control room zone. The air entering each node from the atmosphere was conservatively assumed to be undiluted and of centerline (maximum) chlorine concentrations. This concentration was then diluted in the free volume of each node.

The contaminated control room air intakes were assumed to close before a parcel of contaminated air could reach the control room, but 5 ft^3/min of undiluted outside air was conservatively assumed to enter the control room to account for a "short circuit" leak path from a supply register pointed directly at a control room door. Wear on door seals and aging of penetration seals were also conservatively taken into consideration.

The following additional conservatisms were used in the analysis:

(1) No credit was taken for the removal of chlorine by air that is recirculated through charcoal filters after the essential HVAC units have been actuated.

(2) Less than full credit has been taken for the free volume of air within the nodes and control room zone that is available to dilute incoming air.

(3) A total inleakage flowrate of 750 ft^3/min has been assumed to enter the control room after isolation.

(4) No credit was taken for the physical separation of the control room zone outside air intakes, nor for the fact that they isolate independently.

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(5) Centerline chlorine concentrations were assumed to occur at all control room zone and control building HVAC outside air intakes simultaneously despite the fact that the intakes are physically separated.

(6) No credit was taken for a remote (non-Class 1E) chlorine detector at the nuclear service cooling water chlorine building which annunciates inside the control room zone and which could provide advance warning.

(7) The wind direction was assumed to be directly toward the control room air intakes from the release point.

(8) A positive 1/8-in. water gauge pressure differential was assumed to exist across all leak paths driving contaminated air from the control building into the control room zone following isolation.

The results of this analysis showed that operators would have a minimum of 8 min to don respiratory equipment, which is four times longer than that suggested as a minimum in RG 1.95. The staff has reviewed the applicant's analysis and concludes that the model has conservatively estimated chlorine concentrations in the control room.

The applicant has also committed in a letter dated May 2, 1985, to train operators to distinguish the smell of toxic chemicals and to be able to don breathing apparatus in 2 min.

The staff concludes that the Vogtle control room habitability systems will adequately protect the control room operators in accordance with the requirements of NUREG-0737, Item III.D.3.4, and General Design Criterion (GDC) 19 (Appendix A to 10 CFR 50). Therefore, the staff considers open item 4 regarding toxic gas evaluation of chemicals fully resolved.

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7 INSTRUMENTATION AND CONTROLS

7.3 Engineered Safety Features Systems

7.3.3 Specific Findings

7.3.3.11 Electrical Tunnel Ventilation System

In the SER, the staff stated that the plant's Technical Specifications would require local surveillance of each electrical tunnel every 12 hours because of a concern over a lack of redundancy in the electrical tunnel ventilation system. After the SER was issued, the applicant informed the staff at its January 29-30, 1986, site visit that the Technical Specifications would not include such surveillance. The staff has reviewed the previous information provided by the applicant and concludes that plant Technical Specifications for local surveillance of each tunnel are not required because the electrical cable qualification temperature (194°F) is not exceeded by the maximum temperature (130°F) expected inside the tunnels over a 10-day period without tunnel ventilation system operation.

7.5 Information Systems Important to Safety

7.5.2 Specific Findings

7.5.2.1 Emergency Response Capability--RG 1.97, Revision 2, Requirements

Generic Letter 82-33 requested applicants and licensees to provide a report to the NRC describing how their postaccident monitoring instrumentation meets the guidelines of RG 1.97 as applied to emergency response facilities. In the SER, the staff indicated that the applicant's compliance with RG 1.97, Revision 2, was an open item. The applicant's response to RG 1.97 was provided by letters dated April 14, 1983 (Dutton, Georgia Power Company), March 15, 1985, June 20, 1985, September 6, 1985, and January 21, 1986.

A detailed review and technical evaluation of the applicant's submittals was performed by EG&G Idaho, Inc., under contract to the NRC, with oversight by the NRC staff. This work was reported by EG&G in the Technical Evaluation Report (TER), "Conformance to Regulatory Guide 1.97, Vogtle Electric Generating Plant, Unit Nos. 1 and 2," dated April 1986 (see Appendix M). After this TER was issued, the applicant committed to upgrade its containment sump water temperature instrumentation to conform with the guidance of RG 1.97. The TER, therefore, does not reflect this latest information. Except for the conclusion regarding containment sump water temperature, the staff concurs with the conclusion that the applicant either conforms to, or is justified in deviating from, the guidance of RG 1.97 for each postaccident monitoring variable.

After the generic letter was issued, the NRC held regional meetings in February and March 1983 to answer licensee and applicant questions and concerns regarding the NRC policy on RG 1.97. At these meetings, it was established that the staff

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review would only address exceptions taken to the guidance of RG 1.97. Furthermore, where licensees or applicants explicitly state that instrument systems conform to the provisions of the regulatory guide, no staff review would be necessary for these items. Therefore, the review performed and reported by EG&G only addresses exceptions to the guidance of RG 1.97. This evaluation addresses the applicant's submittals based on the review policy described in the NRC regional meetings and the conclusions of the review as reported by EG&G.

The staff has reviewed the evaluation performed by EG&G contained in the TER and concurs with its bases and findings. The applicant either conforms to, or has acceptably justified deviations from, the guidance of RG 1.97 for each post-accident monitoring variable except for containment sump water temperature.

RG 1.97 recommends that instrumentation be provided to monitor the containment sump water temperature. The applicant originally intended to provide this instrumentation, which conformed to the RG 1.97 criteria for Type D, Category 2 variables except that the electrical connections at the instruments and at the penetrations were not qualified for a steam environment. Residual heat removal (RHR) heat exchanger inlet temperature is an alternate variable that can be used to obtain a measure of the sump water temperature during the recirculation phase of an accident. The applicant however stated that the RHR inlet temperature instruments are not qualified for the high radiation levels that may be present in the heat exchanger rooms. Thus, there was no assurance that either the containment sump water temperature instrumentation or the RHR heat exchanger inlet temperature instrumentation would be available during the course of an accident.

In further justifying the acceptability of this variable, the applicant indicated that the determination of containment emergency sump water temperature is not required because this indication is not used by the operator to take any type of corrective action and because the sump temperature would not rise to the point where the RHR pump net positive suction head would be inadequate. Although the staff agrees with the applicant that emergency operating procedures do not rely on measured values of sump water temperature, the staff believes that this instrumentation is necessary to provide a quantitative measure of heat removal by way of the containment sump and permit monitoring of the containment heat removal system operation and effectiveness.

By letter dated April 25, 1986, the applicant committed to modify containment sump water temperature instrumentation in one loop in each unit to meet Category 2 requirements. The staff finds this commitment acceptable.

On the basis of its review of the TER and the applicant's submittals through April 25, 1986, the staff concludes that the Vogtle design is acceptable with respect to conformance to RG 1.97, Revision 2. Therefore, open item 6 is fully resolved.

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8 ELECTRIC POWER SYSTEMS

8.4 Other Electrical Features and Requirements for Safety

8.4.5 Physical Identification and Independence of Redundant Safety-Related Electrical Systems

In Sections 8.4.5 and 8.4.6 of the SER, the staff stated that it would confirm that circuit breakers in the primary circuit of regulating transformers used as isolated devices are coordinated with their upstream circuit breakers. In Amendment 20 to the FSAR, the applicant states that the primary circuit breaker of the regulating transformer is coordinated with the upstream load center breaker so that the primary circuit breaker will trip before the load center breaker under any fault condition. This resolves the staff's concern on the coordination of these circuit breakers and is, therefore, acceptable to fully resolve confirmatory item 31.

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9 AUXILIARY SYSTEMS

9.1 Fuel Storage and Handling

9.1.2 Spent Fuel Storage

In FSAR Amendment 22, the applicant revised the spent fuel storage rack design The revised design provides a center-to-center spacing of 10.6 in. for the fuel elements and is a free-standing type. In Section 9.1.2 of the SER, the staff provided its evaluation of a spent fuel storage rack having a center-to-center spacing of 13.0 in. and of a type that is attached to the spent fuel pool walls and floor.

The applicant must revise the FSAR to provide the details of the new spent fuel storage rack design, including a criticality analysis, seismic analysis, materials compatibility analysis, and fuel handling accident analysis. The applicant should address the applicable portions of Section 3.8.4 of the Standard Review Plan (SRP, NUREG-0800), Appendix D, regarding seismic analysis; SRP Section 9.1.2 regarding spent fuel rack design; and the appropriate items identified in the letter from B. Grimes, NRC, to all power reactor licensees dated April 14, 1978, "Review and Acceptance of Spent Fuel Storage and Handling Application."

Until the applicant provides the additional information noted above concerning the spent fuel storage rack design, the staff cannot conclude that the revised design meets the acceptance criteria in SRP Section 9.1.2. The spent fuel storage facility will, therefore, be an open item to be addressed in a future supplement.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

This section discusses the staff's evaluation of additional information provided by the applicant in FSAR Amendment 20, an April 8, 1986, meeting, and letters dated September 30, November 11, and December 9, 1985, and January 16 and February 7, 1986.

FSAR Amendment 20 incorporated changes resulting from continued program development. Only those changes that affect the staff's previous evaluation are addressed in this supplement. The applicant's submittals discussed deviations concerning automatic sprinkler protection inside containment, a non-fire-rated hatch cover, a sprinkler system flushing procedure, oversized fire damper assemblies, and fire damper assembly installation. The oversized fire dampers issue was identified as an open item in the SER.

9.5.1.1 Fire Protection Program Requirements

The applicant has incorporated a table of exceptions to the National Fire Protection Association (NFPA) standards into the FSAR by Amendment 20 (Table 9.5.1-9).

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With the exception of one deviation from NFPA 13, which is addressed in Section 9.5.1.5 of this supplement, the exceptions to NFPA standards identified in FSAR Table 9.5.1-9 are not applicable to the nuclear industry in general or to Vogtle in particular and are consistent with staff fire protection guidelines and the plant Technical Specifications, or represent conditions that have been reviewed and approved by the staff in the Vogtle SER and its supplement. The identified exceptions, excluding the deviation from NFPA 13, are, therefore, acceptable.

9.5.1.4 General Plant Guidelines

Building Design

Section C.5.a.(4) of Branch Technical Position (BTP) CMEB 9.5-1 (NUREG-0800) states that penetration openings for ventilation systems should be protected by fire dampers having a rating equivalent to that required of the barrier. During a meeting on April 8, 1986, the applicant informed the staff that the installation of certain wall and floor fire damper assemblies deviates from the manufacturer's installation requirements and the Underwriters Laboratory (UL) tested configurations in that rigid seismic supports have been provided and/or spacers have been installed between the damper sleeves and the barrier openings. These installation deviations negate the UL listing of the damper assemblies and, consequently, the fire resistance rating of the assemblies. The subject dampers do not, therefore, meet Section C.5.a.(4) of BTP CMEB 9.5-1.

To determine the acceptability of these fire damper assemblies as installed, the applicant tested several wall and floor damper assemblies in accordance with the requirements and methods of UL Standard 555, "Standard for Fire Dampers and Ceiling Dampers." Damper assemblies identified by the applicant as being representative samples of worst-case configurations were tested by UL in its laboratories.

The applicant stated that all of the fire damper assemblies satisfactorily withstood the fire endurance test for the full 3-hour test period. Several of the damper assemblies did, however, fail the hose stream test to varying degrees.

To support its position that the subject damper assemblies provide an adequate level of fire safety, the applicant agreed to submit the UL fire test reports and supporting documentation and an analysis of the potential adverse effects on safe shutdown in the event a damper assembly without connecting ductwork should fail in use when subjected to the impact and cooling effects of a hose stream. This is an open item pending staff receipt and review of this material.

In the SER, the staff identified 17 multisectional fire damper assemblies that are not UL listed because their size exceeds the manufacturer's test limitations and, therefore, do not meet Section C.5.a.(4) of BTP CMEB 9.5-1. The staff stated that the applicant should either provide approved fire dampers for all ventilation system penetrations or justify the use of the oversized damper assemblies. The oversized damper issue was identified as an open item in the SER. By letters dated November 11, 1985, and January 16, 1986, the applicant provided additional information. In the January 16, 1986, letter, the applicant stated that, of the 17 oversized fire dampers identified in the SER, only three Unit 1 dampers are now located in fire area boundaries because of the reanalysis of fire area boundary locations. Therefore, there are now only three fire damper assemblies per unit (since Unit 1 and Unit 2 are the same in this regard) that do not meet Section C.5.a.(4) of BTP CMEB 9.5-1.

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These three fire damper assemblies per unit are fabricated of the same materials, using the same methods, and to the same design and procedures as the manufacturer's standard UL-approved 3-hour multiple fire damper assemblies. To obtain a UL listing, the size of a single fire damper in a multiple damper assembly is limited to 36 in. x 36 in. The individual fire dampers that make up the oversized damper assemblies meet this size limitation at Vogtle. However, the oversized fire damper assemblies do not meet the overall size requirements, which are limited to 36 in. x 72 in. for a damper assembly installed in a horizontal configuration and to 72 in. x 72 in. for a damper assembly installed in a vertical configuration.

Fire damper A-1551-S7-612 is a 94-in. x 48-in. vertical damper assembly installed in a barrier separating auxiliary building fire areas 1-AB-L2-A (fire zone 53) and 1-AB-L1-B (fire zone 149). Fire area 1-AB-L1-B contains train B safe shutdown electrical distribution equipment and cables. Fire area 1-AB-L2-A contains train A safe shutdown cables and the train B safe shutdown room cooler that serves fire area 1-AB-L1-B. The train A safe shutdown cables are about 75 ft from the oversized damper assembly and the train B room cooler unit. Both fire areas are equipped with automatic fire detectors and partial-area automatic sprinkler protection. The sprinklers in fire area 1-AB-L1-B are located in the immediate vicinity of the oversized damper assembly. The sprinklers in fire area 1-AB-L2-A protect the intervening space between the oversized damper assembly, the train B room cooler unit, and the train A safe shutdown cables.

Fire damper 1-1551-S7-626 is a 64-in. x 54-in. horizontal damper assembly installed in a barrier separating auxiliary building fire areas 1-AB-L1-B (fire zone 149) and 1-AB-LD-B (fire zone 46). Fire area 1-AB-L1-B contains train B safe shutdown electrical distribution equipment and cables. Fire area 1-AB-LD-B is a large fire area that encompasses several levels in the central part of the auxiliary building. No safe shutdown equipment or cables are located in fire zone 46. Both fire areas are equipped with fire detectors, and fire area 1-AB-L1-B is equipped with partial-area automatic sprinkler protection directly over the oversized fire damper assembly.

Fire damper A-1551-S7-523 is a 52-in. x 44-in. horizontal damper assembly installed in a barrier separating fire areas 1-AB-LD-B (fire zone 46) and 1-AB-LD-A (fire zone 11B). Fire area 1-AB-L1-B is a large fire area encompassing several levels in the central part of the auxiliary building. Level 1 (fire zone 46) contains no safe shutdown equipment or cables. Fire area 1-AB-LD-A (fire zone 11B) also encompasses several levels in the central section of the auxiliary building. Fire area 1-AB-LD-A contains predominantly train B safe shutdown residual heat removal system equipment and train B safe shutdown electrical cables. A major radiation shield wall with a labyrinth entrance is located between the oversized fire damper assembly and the portion of fire area 1-AB-LD-A containing safe shutdown equipment. There are no cable trays or other combustible materials of significance at level 1 of fire area 1-AB-LD-B in the vicinity of the oversized fire damper assembly. Both areas are equipped with automatic fire detectors, and fire area 1-AB-LD-A is equipped with partial-area automatic sprinkler protection.

The staff was concerned that a fire in one of the subject fire areas would spread through one of the oversized fire damper assemblies into the adjoining fire area resulting in damage to redundant safe shutdown systems and impairment of safe shutdown capability. However, because the design, materials,

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fabrication, and installation of the subject damper assemblies are equivalent to those of the manufacturer's standard UL-listed fire damper assemblies, the staff has reasonable assurance that the dampers will operate as intended under fire exposure, thereby preventing the spread of fire from one fire area to another and preserving safe shutdown capability. Morever, if a fire should occur, the staff has reasonable assurance that it would be detected by fire detectors located in the area of origin or by plant operators who would summon the plant fire brigade. Until the arrival of the brigade, the location and arrangement of the combustible materials, the perimeter construction of the area, and/or the heating, ventilation, and air conditioning (HVAC) system duct construction would limit fire spread. If a fire should originate in a location containing a concentration of combustible materials (e.g., cables in trays), the automatic sprinklers would operate and control the fire.

On these bases and because the redundant safe shutdown-related components located in the fire zones around the oversized damper assemblies are well separated, the staff does not expect a loss of safe shutdown capability as a result of anticipated fires even if one of the oversized fire damper assemblies should fail to operate. The staff concludes, therefore, that replacement of the subject oversized fire damper assemblies with UL-listed fire damper assemblies would not significantly increase the level of plant fire safety. The use of the oversized fire damper assemblies identified in the applicant's January 16, 1986, letter is, therefore, an acceptable deviation from Section C.5.a.(4) of BTP CMEB 9.5-1.

Safe Shutdown Capability

Section C.5.b.(2) of BTP CMEB 9.5-1 states that fire barriers with a minimum fire resistance rating of 3 hours should be provided to separate redundant safe shutdown systems. By letters dated November 11, 1985, and January 16, 1986, the applicant identified one non-fire-rated hatch cover installed in a 3-hour-fire-rated barrier. The hatch cover, which is 1/4-in.-thick A36 steel checkered plate, is located at level A of the auxiliary building wing area and separates the train B component cooling water pump room (fire zone 37) from the level B (fire zone 26A) corridor area. This corridor contains a train A safe shutdown raceway. All train A safe shutdown components are located at least 30 horizontal feet from the hatch opening, and all train B safe shutdown components are located at least 5 horizontal feet from the hatch opening. The aggregate horizontal spacial separation between redundant safe shutdown trains is, therefore, at least 35 ft. There are no in situ combustible materials located directly above or below the hatch opening. The only intervening combustible between the redundant trains is a single cable tray located at least 15 ft from the underneath of the hatch opening on level B. Fire detectors are provided on both sides of the hatch, and an automatic preaction sprinkler system is provided both above and below the hatch.

The staff was concerned that a fire in the vicinity of the non-fire-rated hatch cover would spread through the hatch and damage redundant safe shutdown trains. However, because the fuel load in the vicinity of the hatch is negligible, the staff does not expect a fire of significant magnitude or duration to occur. If a fire should occur, the staff would expect it to be detected in its incipient stages by the fire detectors and extinguished by plant operators or the fire brigade before it threatened redundant safe shutdown systems. In the event of rapid fire growth, the staff would expect the automatic sprinklers to operate and control the fire. Moreover, although the hatch cover is not fire rated, it

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is the staff's judgment that it will withstand the hazards present and prevent the spread of fire from one area to another at least until the fire brigade arrives or the sprinkler system operates. In the unlikely event a fire breaches the hatch cover, the lack of significant concentrations of combustible materials around the hatch opening, the relatively large horizontal spacial separation between redundant trains, and the presence of automatic sprinklers in the adjoining area contribute to provide reasonable assurance that the extent of fire spread beyond the hatch will be limited and that redundant safe shutdown systems will not be damaged. On these bases, the staff has concluded that the use of a non-fire-rated hatch cover between fire zones 37 and 26A is an acceptable deviation from Section C.5.b.(2) of BTP CMEB 9.5-1.

The staff will complete its review of the Vogtle safe shutdown capability on receipt of information from the applicant and report its findings in a future supplement to the SER.

9.5.1.5 Fire Detection and Suppression

Sprinkler and Standpipe Systems

Section C.6.c.(3) of BTP CMEB 9.5-1 states that fixed water extinguishing systems should conform to the requirements of NFPA 13, "Standard for the Installation of Sprinkler Systems." Section 1-11.1.1 of NFPA 13 requires that underground mains and lead-in connections to sprinkler system risers be flushed to remove foreign materials that may have entered the underground piping during the course of the installation before connection is made to sprinkler piping. In FSAR Table 9.5.1-9, the applicant stated that the underground main and lead-in piping will not be flushed before connection is made to the sprinkler piping. In FSAR Amendment 23, the applicant provided additional justification for this deviation and referenced the Vogtle flush procedures. The applicant should submit its flush procedures for staff review. This deviation is an open item.

9.5.1.6 Fire Protection of Specific Plant Areas

Containment

Section C.7.a of BTP CMEB 9.5-1 states that fire protection should be provided in containment areas for hazards identified by the fire hazards analysis and that protection should be provided by fixed automatic systems. In the SER, the staff stated that a sprinkler system is provided below the reactor coolant pumps and in areas of high cable concentrations in accordance with Section C.7.a of BTP CMEB 9.5-1. By letter dated September 30, 1985, the applicant informed the staff that a sprinkler system would not be installed inside containment. By letters dated December 9, 1985, and February 7, 1986, the applicant submitted its technical justifications in support of the deviation.

The applicant has justified this deviation on the following bases:

(1) Automatic fire detectors have been provided for the principal fire hazards in the containment.

(2) An oil collection system has been provided for each reactor coolant pump.

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(3) An integral water fire suppression system has been provided for eacn charcoal filter unit.

(4) Cable concentrations, as defined in Section C.5.e of BTP CMEB 9.5-1, are not present.

(5) Redundant safe shutdown components are or will be separated in accordance with staff fire protection guidelines for non-inerted containments through spacial separation, the installation of radiant energy shields, and/or the protection of raceways with fire-rated barriers.

On the basis of the applicant's separation commitment and the staff's review of the applicant's technical justifications and supporting information and the fire hazards analysis for the containment included in Appendix 9A of the FSAR, the staff has concluded that there are no unmitigated fire hazards inside containment that present a credible threat to redundant safe shutdown systems. The staff concludes that the installation of an automatic sprinkler system inside containment would not significantly increase the level of fire safety over that provided by the existing and proposed fire protection and design features. The lack of an automatic fire suppression system is, therefore, an acceptable deviation from Section C.7.a of BTP CMEB 9.5-1.

9.5.1.7 Conclusions

The approved deviations from the staff fire protection guidelines of BTP CMEB 9.5-1 through this supplement to the SER are

- non-UL-labeled oversized fire damper assemblies (SSER 2)
- non-fire-rated hatch cover (SSER 2)
- fire pump controllers not UL listed (SER)
- lack of an automatic water suppression system in the control room peripheral rooms (SER)
- lack of automatic sprinkler protection inside containment (SSER 2)
- lack of smoke detectors in some control room cabinets (SSER 1)
- diesel generator day tank capacity (SER)

The staff will condition the operating license to require that the applicant implement and maintain in effect all provisions of the approved fire protection program.

9.5.2 Communication Systems

9.5.2.2 Interplant (Plant-to-Offsite) Communication Systems

The staff stated in Section 9.5.2 of the SER that interplant communications had not been adequately addressed in the FSAR and that the applicant should upgrade the discussion in FSAR Section 9.5.2 following the guidance in SRP Section 9.5.2. This matter was a confirmatory item.

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In Amendment 22 to the FSAR, the applicant provided the required discussion of the plant-to-offsite communication systems, which consist of

(1) a microwave hotline from the control room to the Georgia Power Company general offices

(2) the Southern Bell Telephone Company system lines

(3) direct extensions from the private automatic branch exchange switchboard located at the general offices in Atlanta

The staff has reviewed the information provided on these systems and finds that the guidelines of SRP Section 9.5.2 have been met. Therefore, the staff concludes that confirmatory item 34 concerning plant-to-offsite communication systems is satisfactorily resolved.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.5.4.2 Emergency Diesel Engine Fuel Oil Storage and Transfer System

As stated in the SER, the staff found the design of fuel oil storage and transfer systems acceptable pending confirmation that procedures have been developed and implemented which require that on the fifth day (or sooner) of continuous diesel generator operation, additional fuel will be ordered and all manual actions necessary for receiving fuel oil have been performed before receipt of the fuel oil.

In Amendments 22 and 23 to the FSAR, the applicant stated that administrative controls and plant procedures have been developed and implemented which will require that fuel oil be ordered after 5 days of continuous diesel generator operation, the completion of manual actions necessary to receive the fuel oil, and ample time for sediment in the tank to settle before the fuel oil is used. On the basis of post-loss-of-coolant-accident loads and the most adverse environmental conditions, the staff finds that there is sufficient time for delivery of the fuel oil to the site, the performance of the necessary manual actions to receive the fuel oil, and a 24-hour settlement time for the filled fuel oil storage tank. Therefore, the staff concludes that confirmatory item 37 regarding procedures for replenishment of the diesel engine fuel oil supply has been satisfactorily resolved.

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

In Section 9.5.8 of the SER, the staff stated that, with regard to dust and dirt on diesel generator (DG) electrical components, the applicant committed to clean control panels before plant startup and as a part of general housecleaning and maintenance activities at regular intervals thereafter. The staff indicated that it would find the applicant's design of the DG control panels acceptable on confirmation that procedures for general housecleaning and maintenance activities have been implemented and the term "regular intervals" has been quantified. This issue was a confirmatory item.

In Amendment 22 to the FSAR, the applicant stated that housecleaning and maintenance procedures had been developed which will require cleaning of the control

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panels every 18 months to prevent the accumulation of dust on electrical components. The staff finds the applicant's approach to dust and dirt control in the DG area acceptable. Therefore, the staff concludes that confirmatory item 38 concerning procedures for general housecleaning and maintenance activities (at prescribed intervals) is satisfactorily resolved.

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13 CONDUCT OF OPERATIONS

13.2 Training

In the SER, the staff described the applicant's training program for Vogtle licensed reactor operators and nonlicensed plant staff and concluded that, as described, the program was acceptable. However, by letter dated April 4, 1986, the applicant proposed substantive changes to its training program. The staff has reviewed this submittal and its evaluation follows.

13.2.1 Licensed Operator Training Program

13.2.1.1 Initial Training Program

The applicant has deleted previously incorporated FSAR Sections 13.2.1.1.3 and 13.2.1.1.4, which discussed Vogtle license and Technical Specifications, and fuel handling and core alterations training, respectively, for licensed operators. These required segments of training are still listed in FSAR Tables 13.2.1-1 through 13.2.1-5 as part of the licensed reactor operator and senior reactor operator (SRO) syllabuses. The training on the Vogtle license and Technical Specifications has been incorporated as part of the control room operations training, and the fuel handling and core alterations training has been incorporated into Vogtle systems training without any changes to the duration of these augmented training segments. The applicant should clarify the intent of these deletions and the scope of these training segments because this change appears to significantly reduce the duration and scope of these segments of training.

The inspection of the applicant's training programs by the staff, as documented in Office of Inspection and Enforcement (IE) Inspection Report 50-424/85-51, revealed that the training for the mitigating core damage program is of insufficient scope and duration (i.e., 8 hours) to cover the topics required by NUREG-0737. Further, the applicant's program of training for mitigating core damage, which appears to be primarily a self-study course, is not comparable to 40-hour lecture courses that cover problem solving and calculations, simulator training, and use of the process computer. The applicant should demonstrate that the scope of the program of training for mitigating core damage is appropriate. Therefore, mitigating core damage training (NUREG-0737, Item I.A.2.1) is an open item.

The applicant has reduced the duration of nuclear power plant theory training in licensed SRO training described in FSAR Tables 13.2.1-3 and 13.2.1-4. The applicant should clarify the nature and extent of the reduced scope of nuclear power plant theory training, particularly for required training in heat transfer, fluid flow, and thermodynamics. The review and audit segment of licensed SRO training in Table 13.2.1-3 has been deleted. The applicant should correct this omission.

The duration of control room operations training for licensed SRO trainees with no previous nuclear power plant experience in FSAR Table 13.2.1-5 is less than

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that specified for licensed reactor operator trainees shown in FSAR Table 13.2.1-6, while including more training, for example, on the Vogtle license and Technical Specifications. The applicant should clarify the scope of this segment of licensed SRO training to describe how more training can be provided in a shorter duration for the license candidates with no prior nuclear power plant experience. Licensed operator training is an open item.

13.2.1.2 Licensed Operator Requalification Training Program

The applicant has stated that the performance of licensed operators will be evaluated annually on the Vogtle simulator. Certified simulator instructors are to be exempt from this requirement. Although exemptions from portions of the licensed operator requalification program are granted on a limited basis to instructors, assignments should be rotated annually so that all licensed and certified instructors successfully complete a full requalification program in each 2-year requalification cycle. The applicant should modify FSAR Section 13.2.1.3.2.3 to reflect limited use of waivers for annual simulator performance evaluations.

The applicant's instructor requalification program as described in FSAR Section 13.2.1.3.3 permits instructors to complete requalification training through self-study and examination only. Although licensed or certified instructors may be exempt from portions of the requalification training lectures which they ordinarily teach, they may not be exempt from the entire requalification program, nor is individual study considered an acceptable substitute for a requalification lecture series per 10 CFR 55, Appendix A. Licensed or certified contract instructors who teach systems, integrated response, transients, and simulator courses are similarly required to participate fully in a requalification program that meets the requirements of 10 CFR 55, Appendix A. The staff finds the applicant's description of instructor participation in requalification training to be unacceptable and, therefore, considers it an open item.

The applicant's description of instructor qualification in Section 13.2.1.3.3 of the FSAR implies that only new instructors who teach systems, integrated response, transients, and simulator courses to licensed operators must complete an SRO examination. The applicant should provide clarification that all instructors, including contractors, of systems, integrated response, transients, and simulator courses be licensed or certified. Instructor qualification is an open item.

13.2.2 Training for Nonlicensed Plant Staff

The applicant has stated in FSAR Section 13.2.2 that nonlicensed personnel may be trained to perform specific tasks if they do not meet the education and experience requirements of Standard N18.1-1971 of the American National Standards Institute (ANSI). Descriptions of training programs for nonlicensed personnel state that, following fuel load and initial criticality, initial training for nonlicensed personnel will be performed and such training normally will be completed within 1 year of assignment to a job position. The staff considers that initial training for nonlicensed personnel should be provided to prepare personnel to perform job duties, i.e., before preoperational tests, before fuel

loading, or before appointment or reappointment to the position. The staff's position is consistent with SRP acceptance criteria (Section 13.2.2.II). The applicant should clarify that initial training will be provided for nonlicensed personnel before they are assigned to perform job duties independently.

The numbers and types of nonlicensed personnel receiving training should be indicated by referencing the appropriate figure (not table) in the FSAR text.

Initial training for foremen and supervisors in mechanical maintenance, electrical maintenance, health physics, chemistry, and instrumentation and control is not discussed in the applicant's revised FSAR. The applicant should revise the FSAR to include initial training for foremen and supervisors in preparation for independent job duties.

Initial training for nonlicensed operators before they are assigned to independent plant equipment manipulations has been deleted by proposed changes to FSAR Section 13.2.2.1.7. According to the applicant's program description, nonlicensed operators will complete initial training within 1 year of assignment to the position or may be trained to perform specific tasks if they do not meet ANSI N18.1-1971 qualifications. The staff finds that the applicant's training for nonlicensed operators, as described, is unacceptable in that it is not designed to prepare personnel for their job duties before their appointment to the job position. The applicant must clarify that training to perform specific tasks will be accomplished before personnel undertake these tasks. Nonlicensed personnel training is an open item.

The applicant has not described provisions for retention of records of the qualifications, experience, training, and retraining of each member of the plant organization as specified by ANSI N18.1-1971. Failure to maintain records was also identified by a staff inspection of Vogtle training programs (Inspection Report 50-424/85-51). The applicant should commit in the FSAR to retain all plant personnel training records, including waivers of required training, for the life of the plant in order to be consistent with the proposed Vogtle Technical Specifications. Provisions for retention of training records is an open item.

13.3 Emergency Planning

13.3.1 Introduction

In the SER, the staff identified emergency planning as an open item. By letter dated July 29, 1985, the staff transmitted a draft evaluation of the Vogtle Emergency Plan (EP) to the applicant that provided details of staff questions that would have to be resolved satisfactorily to close the open item. The staff's review was through Revision 1 of the EP dated May 17, 1985. Since issuance of this evaluation, the applicant has submitted Revisions 2 through 5 dated November 15, 1985, December 31, 1985, February 14, 1986, and February 25, 1986. The applicant has also provided public information materials for Georgia and South Carolina by letter dated April 22, 1986. Additionally, the staff met with the applicant on October 16, 1985, and December 11, 1985, to discuss the Vogtle EP. This evaluation is based on the latest applicant submittals. The acceptance criteria used as a basis for the staff's review of the Vogtle EP are specified in SRP Section 13.3, "Emergency Planning," and include the planning standards of

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10 CFR 50.47(b), the requirements of Appendix E to 10 CFR 50, and the specific criteria of NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," dated November 1980. The criteria of NUREG-0654 have been endorsed in RG 1.101, Revision 2, "Emergency Planning and Preparedness for Nuclear Power Reactors," dated October 1981 and thus have the same status as a regulatory guide.

Evaluation of the state of emergency preparedness for Vogtle also involves the review of State and local radiological emergency response plans by the Federal Emergency Management Agency (FEMA). The SRP states that the FEMA findings on offsite plans are reviewed by the NRC and a full-scale exercise is to be conducted at the facility. In accordance with the revised rule on emergency planning (47 FR 30232), no NRC or FEMA findings and determinations concerning the state or adequacy of offsite emergency preparedness are required before issuance of an operating license authorizing only fuel loading and low-power operations up to 5% of rated power. The findings and determinations of FEMA on the adequacy of the State and local emergency response plans and the overall conclusion of the NRC on the state of emergency preparedness for Vogtle will be provided in a future supplement to the SER.

Section 13.3.2 of this supplement lists each planning standard of 10 CFR 50.47(b) followed by an evaluation of the applicable portions of the Vogtle EP that principally relate to that particular standard. Section 13.3.3 of this supplement provides the staff's conclusions.

13.3.2 Evaluation of the Emergency Plan

13.3.2.1 Assignment of Responsibility (Organizational Control)

Standard

Primary responsibilities for emergency response by the nuclear facility licensee and by the State and local organizations within the Emergency Planning Zones (EPZs) have been assigned, the emergency responsibilities of the various supporting organizations have been specifically established, and each principal response organization has staff to respond and to augment its initial response on a continuous basis.

Emergency Plan Evaluation

In the event of an accident at Vogtle, the applicant's staff is responsible for recognizing and declaring an emergency condition, classifying the event, notifying and activating the onsite organization, notifying offsite authorities, taking corrective actions to mitigate accident consequences, requesting additional support if needed, establishing and maintaining effective communications with onsite and offsite organizations, taking protective actions on site, recommending protective actions off site, monitoring and controlling radiation exposures of applicant personnel, and providing public information on the event in conjunction with State and local authorities. The applicant's emergency response is carried out under the control of the Emergency Director. Initially the On-Shift Operations Supervisor assumes the position of Emergency Director until properly relieved by the General Manager-Vogtle Nuclear Operations or

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one of the following: Senior Vice President-Nuclear Operations or the Vice President and General Manager-Nuclear Operations from the applicant's headquarters, or the Vogtle Manager-Unit Operations or the Vogtle Superintendent of Operations.

Vogtle is located in Burke County, Georgia, near the Savannah River across from the Savannah River Plant on the South Carolina side of the river. The Savannah River Plant is operated by the U.S. Department of Energy (DOE) and covers most of the area within the plume exposure pathway EPZ in the State of South Carolina except for small areas of Aiken, Barnwell, and Allendale Counties. The plume exposure pathway EPZ within the State of Georgia is contained within Burke County and excludes a small uninhabited area of Richmond County as well as several small areas of Burke County within 10 miles of the Vogtle plant.

Since the Savannah River Plant consists of land owned or leased by the Federal Government, DOE is responsible for the direction and control of all emergency response actions within this site. By memorandum of agreement between DOE and the applicant (provided in Appendix 5 of the EP), it is agreed that they will provide a 24-hour point of contact with each other, will maintain lines of communication, and will promptly notify each other of emergencies as well as exchange meteorological and radiological information. In the event of an emergency at Vogtle, DOE will promptly notify all personnel at the Savannah River Plant within the plume exposure pathway EPZ, assess the radiological hazards within the Savannah River Plant, and implement any protective actions decided on to protect the health and safety of individuals within the site. The Savannah River Plant will also provide radiological monitoring, assessment, medical support, and dose projection assistance (based on Vogtle source term information and Savannah River Plant meteorology); coordinate public affairs activities; maintain a 24-hour communications capability; and deploy liaison support to the State of South Carolina as described in an agreement between DOE and South Carolina contained in Appendix 2 of the Vogtle EP. However, this agreement is not specifically referenced, nor is this assistance described anywhere else in the Vogtle EP. In addition, the agreement describes the general protective actions that will be implemented by the Savannah River Plant on its site that are not described or referenced in the body of the EP.

The Office of the Governor of South Carolina provides direction, control, and guidance, including protective actions during emergencies. This office also provides an official representative for public information for the State and activates the South Carolina Comprehensive Emergency Preparedness Plan when emergencies are declared. This plan activates the Emergency Preparedness Division (EPD) of the Office of the Adjutant General and the Bureau of Radiological Health (BRH) of the Department of Health and Environmental Control, the two major State agencies involved in radiological emergencies. The Director, EPD, is responsible for coordinating emergency planning of all State, county, and municipal agencies and is the principal advisor to the Governor for emergency response. The Chief, BRH, has the responsibility for providing technical assistance and resources necessary to evaluate and assess radiological accidents as well as providing protective action guidance and radiation monitoring and analyses.

Other South Carolina agencies that provide support during radiological emergencies include the Department of Highways and Public Transportation, which provides traffic control, a primary warning point, communications, heavy

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equipment, and transportation; the Law Enforcement Division, which coordinates law enforcement and search and rescue activities; secures waterways and evacuated areas; and provides communications; the Forestry Commission, which provides communications, transportation, and fire control; the Department of Wildlife and Marine Resources, which provides search and rescue, transportation, public warning, and law enforcement assistance; the Department of Social Services, which coordinates shelter operations and provides emergency welfare, public information, and assistance to families; the Clemson University Cooperative Extension Service, which provides agricultural data for radiological assessment and locating contaminated food products; and the Department of Agriculture, which impounds contaminated food products, coordinates with Federal agriculture authorities, and assists in the disposal and assessment of contaminated food products.

The South Carolina EPD is responsible for establishing and directing the State Emergency Operations Center (EOC) in Columbia as well as the forward EOC for Vogtle located in the National Guard Armory in Graniteville, South Carolina. EPD also coordinates offsite support from Federal, State, and local agencies; determines and directs protective actions; and provides a 24-hour notification capacity. The Vogtle warning point is a South Carolina Highway Patrol dispatcher who notifies the EPD and BRH 24-hour duty officers. EPD contacts the counties to ensure that they have received the notification and contacts the Governor's office for direction. BRH contacts Vogtle to verify the information received from the warning point. EPD alerts other supporting State agencies as necessary; receives radiological assessments and recommended protective actions from BRH, which are passed on to the Governor or his designated representative; and coordinates with the State of Georgia, the Savannah River Plant, and other offsite agencies.

The Georgia Department of Natural Resources (DNR) is the lead agency in responding to peacetime radiological emergencies within the State. DNR is responsible for assessing radiological conditions; providing radiation control, personnel, equipment, and technical expertise on a 24-hour basis; and advising the Governor on whether an emergency exists. Upon being advised of a radiological emergency, the Governor activates the Georgia Emergency Management Agency (GEMA), which is an agency within the Georgia Department of Defense. GEMA is responsible for overall control and coordination of disasters and automatically activates eight other Georgia State agencies that support the emergency effort through the State EOC. The principal support agencies include the Georgia Department of Human Resources, which provides radiation monitoring teams and radiological health expertise; the Georgia Department of Defense, with broad legal authority to protect the health and safety of the public, which provides emergency communications, radiological monitoring assistance, public warning and evacuation procedures, area security and control assistance, and transportation; and the Georgia Department of Public Safety, which provides communications, transportation, and law enforcement functions.

The Georgia Department of Defense will establish a State EOC or a mobile EOC to maintain a 24-hour radio network, commercial telephone, national warning, national communications, teletype, and other communications systems. This center will be located at the Burke County EOC for emergencies involving Vogtle and will be used by GEMA to direct the Georgia emergency response. The DNR Radiation Emergency Coordinator interacts with appropriate State, local, and

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Federal agencies as well as private organizations to direct radiological monitoring and control actions off site. This individual will be located in the State EOC or mobile EOC to carry out his functions when either of these facilities is activated.

Within Burke County, the Chairman of the Board of Commissioners has the overall responsibility for providing radiological emergency response planning, initiating actions, and providing direction and control to county agencies. The Burke County Emergency Management Agency and the Sheriff's Department have the major role in responding to accidents at Vogtle. The Burke County Emergency Management Agency is responsible for receiving emergency notifications from Vogtle and GEMA as well as maintaining communications with them, activating the public notification system, coordinating the Burke County radiological emergency response, activating and directing offsite reception centers, and providing information to the public and the media. The Burke County Sheriff's Office is responsible for providing traffic control and access control within the plume exposure pathway EPZ and backup and ancillary public notification capability and is the alternate for notification from Vogtle and GEMA. The Vogtle EP states that the facilities for emergency notification communications with the Burke County Emergency Management Agency are manned 24 hours per day as is the sheriff's office.

The emergency organizations for Aiken, Barnwell, and Allendale Counties in South Carolina are similar in that the responsibility of overall radiological emergency response resides with the Chairman of the County Council, who must initiate actions and provide direction and control of the county emergency organizations. Each county has an emergency (or disaster) preparedness agency that, in conjunction with the county sheriff's office, is the principal responder to radiological emergencies. It is noted that the exact titles of these county emergency agencies are inconsistent between the written text (page A-17), Table A-1, and Figure A-5 of the Vogtle EP. The county emergency agencies in all three counties are responsible for activating the other responding agencies, the county EOC, and the public notification system in their respective counties. Also, they are responsible for maintaining communications with EPD and Vogtle, providing information to other responding county agencies, activating and directing the operation of county reception centers, and implementing protective actions as directed by EPD or the Chairman of the County Council.

The Sheriff's Department in Barnwell County and the law enforcement communication centers in Aiken and Allendale Counties are the primary points for notification of an emergency at Vogtle. The sheriff in each county is responsible for coordinating traffic control and law enforcement in the event of an evacuation, providing security to the county EOC, controlling access to the plume exposure pathway EPZ, and providing backup public notification.

Agreements for medical support to Vogtle with the Burke County Ambulance Service; the Burke County Hospital in Waynesboro, Georgia; the Humana Hospital in Augusta, Georgia; the Radiation Management Corporation; and six physicians in the Waynesboro-Augusta area are provided in Appendix 2 of the Vogtle EP. In addition, the Waynesboro Fire Department has agreed to provide fire fighting assistance. Arrangements have been made for engineering, technical, and administrative support with the Westinghouse Electric Corporation and the Bechtel Power Corporation. DOE has agreed to provide radiological assessment

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support, and Vogtle is a member of the Nuclear Power Plant Voluntary Assistance Group sponsored by the Institute of Nuclear Power Operations.

The following items require resolution:

- The assistance that DOE agrees to provide to the State of South Carolina as well as a reference to this agreement should be included in the text of the EP.
- The exact titles of the emergency agencies for Aiken, Allendale, and Barnwell Counties should be determined and the appropriate corrections made on page A-17, Table A-1, and Figure A-5 of the EP.

13.3.2.2 Onsite Emergency Organization

Standard

On-shift facility licensee responsibilities for emergency responses are unambiguously defined, adequate staffing to provide initial facility accident response in key functional areas is maintained at all times, timely augmentation of response capabilities is available, and the interfaces among various onsite activities and offsite support and response activities are specified.

Emergency Plan Evaluation

When an emergency or incident occurs, the initial onsite staffing of the Vogtle emergency organization will consist of personnel normally employed at the site. If needed, personnel from other Georgia Power Company organizations (company headquarters, Edwin I. Hatch Power Plant), as well as personnel from other contract or agreement organizations, will augment this staff. Initially, the On-Shift Operations Supervisor, or the Shift Supervisor if the On-Shift Operations Supervisor is not immediately available, will fill the position of Emergency Director and has the authority and responsibility for overall direction of the Vogtle emergency organization. This individual remains the Emergency Director until a more senior member of the Vogtle or Georgia Power Company management arrives at the site and assumes this responsibility. An onshift emergency organization is established, and emergency assignments for the normal operating crew have been made. The relationship between the emergency organization and the shift complement is illustrated in Tables B-1 and B-2 and Figures B-1, B-2, B-3, and B-4 of the Vogtle EP. The positions and/or titles and major tasks and responsibilities for all key personnel in the emergency organization are specified, including the responsibilities that may not be delegated by the Emergency Director.

The applicant's concept of shift augmentation generally adheres to the guidance provided in Table 2 of Supplement 1 to NUREG-0737. There is no 30-min augmentation goal because all of the staffing requirements for the normal shift complement and the 30-min augmentation are filled by the onshift operating crew.

Table B-2 of the Vogtle EP indicates that the General Manager-Vogtle Nuclear Operations, the Senior Vice President-Nuclear Operations, and the Vice President and General Manager-Nuclear Operations are the three primary individuals assigned to the position of Emergency Director. The Vogtle Manager-Unit

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Operations, the Supervisor of Operations, the On-Shift Operations Supervisor, and the Shift Supervisor are listed as alternates for the Emergency Director. The qualifications of these individuals are provided in the Vogtle FSAR, and their emergency response training is given in Section O of the Vogtle EP. The EP states that the On-Shift Operations Supervisor will initially assume the position until relieved by the General Manager-Vogtle Nuclear Operations or one of the other primary designees or one of the higher level supervisors who are alternates after they have been appropriately briefed on the status of the emergency and have become familiar with plant conditions and status. A description of the turnover briefing and procedure is provided, including references to the actual emergency preparedness implementing procedures (EPIPs) governing the turnover.

Appendix 7 of the Vogtle EP provides a copy of the Georgia Power Company Corporate Emergency Plan delineating the responsibilities of the corporate staff in supporting Vogtle in the event of an emergency. This support includes sending the Senior Vice President-Nuclear Operations or the Vice President and General Manager-Nuclear Operations to the site to assume the position of Emergency Director, as well as establishing an operations center at corporate headquarters to assist the site personnel in providing communications to offsite authorities, obtaining emergency resources and engineering support, and providing public information functions. If the Senior Vice President-Nuclear Operations goes to the site, members of the public information staff will accompany him and establish an emergency news center in Waynesboro as well as provide public information support at the Vogtle Emergency Operations Facility (EOF). The responsibilities of the corporate public information staff and how it interfaces with both the site and the nuclear operations corporate staff is described in Appendix 8 of the EP. It is noted that the alternates for the position of Public Information Manager who relocates to the EOF with the Senior Vice President-Nuclear Operations are different individuals in Appendices 7 and 8 than those named as alternates in Table B-2.

The interfaces among the functional areas of the onsite emergency organization and between the onsite and offsite support organizations are provided. The written agreements between the applicant and these support organizations and individuals to provide for public notification; ambulance service, medical, law enforcement, technical, engineering, and fire fighting assistance are identified, and copies of these agreements are appended to the EP.

The following item requires resolution:

- The alternates for the position of Public Information Manager should be determined and their identification made consistent in Table B-2 and Appendices 7 and 8 of the EP.

13.3.2.3 Emergency Response Support and Resources

Standard

Arrangements for requesting and effectively using assistance resources have been made, arrangements to accommodate state and local staff at the licensee's near-site EOF have been made, and other organizations capable of augmenting the planned response have been identified.

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Emergency Plan Evaluation

The Vogtle EP identifies the Federal, State, and local government organizations, private organizations, local support groups, and individuals that can be relied on for assistance in an emergency. Letters of agreement documenting the availability of assistance from the following are provided in Appendix 2 to the EP:

(1) Board of Commissioners of Burke County (ambulance services)
(2) Burke County Sheriff's Department
(3) City of Waynesboro (fire fighting assistance)
(4) Radiation Management Corporation (medical consultation and training)
(5) Humana Hospital
(6) Ralph W. Buchanan, M.D.
(7) Alton F. Garrison, M.D.
(8) Plastic Reconstructive and Cosmetic Associates (medical treatment)
(9) Burke County Hospital
(10) Medical Specialists, Incorporated
(11) Department of Energy, Savannah River Operations Office
(12) Westinghouse Electric Corporation
(13) Bechtel Power Corporation

In addition, there are agreements between (1) DOE and South Carolina and (2) Humana Hospital and South Carolina and Aiken, Allendale, and Barnwell Counties.

The applicant has entered into the Nuclear Power Plant Emergency Response Voluntary Assistance Agreement sponsored by the Institute of Nuclear Power Operations, a copy of which is provided as Appendix B of the Corporate Emergency Plan appended to the Vogtle EP as Appendix 7. In addition, the applicant can obtain engineering, health physics, construction, and general emergency support through Southern Company Services, Inc., and Alabama Power Company, as well as from the Edwin I. Hatch Power Plant, which is also operated by the applicant.

Federal assistance is available from DOE, NRC, FEMA, and the Environmental Protection Agency (EPA). The EP provides a general description of the accommodations available to Federal employees near Vogtle and in Table C-1 gives an extensive list of characteristics for the four largest airports within 35 miles of the site. The anticipated arrival times for DOE, NRC, and FEMA personnel who will provide assistance are given in the EP.

Provisions are made to accommodate representatives from FEMA and NRC as well as the States of Georgia and South Carolina in the Vogtle EOF. In addition, space for NRC representatives is available in the Technical Support Center (TSC). The applicant will send technical representatives to offsite governmental centers as required or requested.

Onsite laboratory facilities are provided for radioactive sample analysis, and a postaccident sampling system (PASS) is available to take reactor coolant and containment sump liquid samples as well as containment atmospheric samples, which are analyzed by a remote control panel and an on-line gamma spectrometer as well as ion and gas chromatographs. The PASS also can be used to obtain grab samples for laboratory analysis. The Vogtle onsite laboratory can provide gamma spectroscopy of diluted samples (1:1000), and the EP states that high

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radioactivity level samples will be sent to Oak Ridge National Laboratory for analysis if the remote PASS is inoperable. However, the EP does not provide for any method of agreement with Oak Ridge to perform this analysis. This function appears to be based on an ad hoc arrangement.

Field monitoring samples and environmental samples obtained during an emergency at Vogtle will be analyzed at onsite laboratories or the Georgia Power Company Environmental Laboratory at Smyrna, Georgia. If the volume of samples or the availability of counting equipment results in the need for additional counting capability, the analysis laboratory at Hatch will also be utilized. All of these laboratories have the capability to perform gamma spectroscopy and gross beta and alpha analysis.

The following item requires resolution:

- The EP should provide a brief description of the physical and administrative means of obtaining PASS sample analysis at Oak Ridge National Laboratory, and a copy of the agreement to perform this analysis should be appended to the EP.

13.3.2.4 Emergency Classification System

Standard

A standard emergency classification and action level scheme, the bases of which include facility system and effluent parameters, is in use by the nuclear facility licensee, and state and local response plans call for reliance on information provided by facility licensees for determination of minimum response measures.

Emergency Plan Evaluation

The emergency classification and action level scheme uses a symptomatic analysis of plant conditions, the indicated or anticipated integrity of fission product barriers, and the presence of certain initiating conditions to determine which of the four emergency classes should be used to classify the event. The four emergency classes are the standard classifications for emergencies: (1) Notification of an Unusual Event (NUE), (2) Alert, (3) Site Area Emergency, and (4) General Emergency. The NUE and Alert classes give early notification of minor events that can develop into more serious ones or that indicate more serious conditions that are not fully developed. A Site Area Emergency reflects conditions that indicate that significant releases of radioactivity are occurring or are likely to occur, but core melt is not imminent. A General Emergency involves actual or imminent core degradation or melting with the potential of an offsite release.

The emergency classes are used to determine the initial steps to be taken by the applicant and are used by offsite authorities to determine the level of preplanned actions to be taken by their emergency organizations. Subsequent actions by offsite authorities are based on projected or actual plant conditions as determined by the Vogtle emergency staff. These projected plant conditions are determined by the applicant on the basis of the status of the reactor core, core cooling systems, radioactive effluent releases, and containment integrity or other parameters.

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Section IV.B of Appendix E to 10 CFR 50 requires that emergency action levels (EALs) be discussed and agreed on by the applicant and State and local authorities. In addition, these EALs must be reviewed by all parties on an annual basis. Although the Vogtle EP indicates that these EALs are a part of the State and local plans and that plant status information will be provided to State and local authorities, there is no definitive statement with regard to an agreement by State and local authorities or any commitment to an annual review.

The following item requires resolution:

- The EP should indicate that the EALs are agreed on by the applicant and State and local authorities and that the EALs will be reviewed annually.

13.3.2.5 Notification Methods and Procedures

Standard

Procedures have been established for notification, by the licensee, of State and local response organizations and for notification of emergency personnel by all response organizations; the content of initial and followup messages to response organizations and the public has been established; and means to provide early notification and clear instructions to the populace within the plume exposure pathway Emergency Planning Zone have been established.

Emergency Plan Evaluation

Procedures are established for notifying onsite and offsite emergency personnel and Federal, State, and local government response organizations. The Emergency Director is responsible for classifying an event and notifying onsite personnel through the plant public address system and the use of tone signals. Separate signals exist for an Alert, Site Area Emergency, and General Emergency as well as a fire, which will be followed by announcements on the public address system. Notification of an Unusual Event will be announced on the public address system without a tone signal. The plant operations duty officer, on duty 24 hours per day, is responsible for notifying plant management, the corporate duty officer, and the Hatch duty officer. Selected plant management also can be notified by a beeper system. The Emergency Director is responsible for notifying the Shift Supervisor of the Vogtle Security Department, who will activate the tone signal. The Security Department is also responsible for notifying all visitors, the Visitor's Center, the Vogtle recreational park, Allen B. Wilson Combustion Turbine Plant, and any individuals located within the site boundary. Backup onsite communications are provided by plant radio services. During normal working hours, management will assemble the emergency staff in the Vogtle Emergency Response Facilities (ERFs). During the backshift, the plant operations duty officer is responsible for notifying Vogtle emergency personnel using a call fanout system described in the EP. The Vogtle Security Department is responsible for notifying and ensuring that all visitors and nonessential personnel are evacuated from the site, the Visitor's Center, the recreational park, and the Wilson plant. Visitors will receive training in the required emergency actions or will be escorted by trained plant personnel.

Notification of the State and local response personnel is the responsibility of the Emergency Director who will use a dedicated telephone system called the

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Emergency Notification Network (ENN). Notification of GEMA, Burke County Emergency Operations Center, South Carolina Highway Patrol, Aiken and Allendale County law enforcement centers, the Barnwell County Sheriff's Office, and the Savannah River Plant Emergency Center is made via the ENN, which is manned 24 hours per day. Declaration of an emergency to these locations can be made within 15 min. These agencies are responsible for notifying the appropriate emergency personnel. Backup communication with these warning points is provided by commercial telephone and radio. The Emergency Director is also responsible for contacting the NRC Operations Center on the emergency notification system (ENS) telephone.

Examples of notification forms for the ENN and the NRC ENS as well as followup message forms are provided in the EP. All notifications by dedicated telephone will be verified by roll call and notifications by radio or commercial telephone by call back.

Public notification for the plume exposure pathway EPZ, except for the area covered by the Savannah River Plant is accomplished by a combination of systems that have the capability to complete initial notification of residences within the EPZ in 15 min. The Savannah River Plant uses its own site notification system to warn employees, visitors, and transients. Followup messages are provided through the emergency broadcast system (EBS).

The initial alerting and notification of the public within the plume exposure pathway EPZ is made primarily by both automatically actuated tone-alert radios activated by the National Weather Service (NWS) and a system of fixed sirens. Anyone not notified by these two systems will be notified in Burke County by vehicles equipped with sirens and/or loudspeakers from the Sheriff's Department, the County Emergency Management Agency, or the Georgia State Patrol. Boats equipped with a voice or sound device will be used to notify sportsmen and other persons engaged in recreational activities on the Savannah River. These boats will be supplied by the Burke County Emergency Management Agency or the Georgia Department of Natural Resources. Door-to-door contact may be used in areas of imminent radiological danger and may be carried out in conjunction with the confirmation of an evacuation.

The tone-alert radio system is activated on a 24-hour-per-day basis by NWS in Augusta, Georgia, at the request of GEMA or the Burke County Emergency Management Agency. The radios will be distributed by the applicant to all residences, schools, and businesses on the basis of electrical service in Burke County. Information regarding radio distribution will be updated monthly using requests for new and disconnected electrical service, and an annual survey of residents within the plume exposure pathway EPZ will be made to ensure that the list is current. Because of the small number of South Carolina residents within the EPZ, information regarding radio distribution to these residents will be updated only by house-to-house or phone survey. The applicant will also determine the residences within Burke County that do not have electrical service and update this list annually. This list will be used by the county to make direct contact during emergencies.

A brochure will be distributed with each tone-alert radio explaining how the radio is to be used, where to locate it, and how it and the backup battery power supply operate, with instructions to return it to the Georgia Power

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Waynesboro District Office if it is defective. NWS will run a weekly test of the tone-alert radio system, and the applicant will test and evaluate it at least once a year.

A system of fixed, rotating sirens will be installed in the plume exposure pathway EPZ and will be sounded in coordination with the tone-alert radios. These sirens will be controlled by the county in which they are located and will be activated at the request of the Chairman of the County Council, the South Carolina EPD, or GEMA. The EP does not provide a physical description of the siren system, where the sirens will be located, the design basis for the system, or the communications systems used to sound them. The EP does not describe the responsibilities and procedures for testing and maintaining the siren system.

The Vogtle EBS consists of 20 radio and 4 television stations, which have agreed to broadcast emergency instructions and information. These agreements are provided in Appendix 2 of the EP. Although most of these stations do not broadcast 24 hours per day, they expect to be ready to broadcast within approximately 30 min after being notified. The EBS can be activated by contacting any of the broadcast stations that are members, although the control station is WRDW-TV 12 in Augusta, which broadcasts 24 hours per day. GEMA, EPD, or any of the counties have the authority to activate the EBS by telephone.

The boats equipped with sirens and/or loudspeakers will be dispatched by the Burke County EOC and are located in Waynesboro. At the Alert emergency class or higher, the boats will be transported to the boat ramp at Vogtle. When ordered, they will be launched and sent upstream and downstream to warn transients along the river. This capability is available 24 hours per day. The activation of the boats will be coordinated with the South Carolina Department of Wildlife.

The vehicles equipped with sirens and/or loudspeakers will be dispatched by the respective counties or by GEMA and EPD. Because the vehicles involved are police or emergency vehicles, they are available on a 24-hour-per-day basis. These vehicles will also be used to provide door-to-door contact of those residents who may not have heard the warning notification.

The following items require resolution:

- The design basis and physical description of the siren warning system, including a map illustrating siren locations and area of coverage, should be appended to the EP.

- The EP should describe the responsibilities and procedures for testing and maintaining the siren warning system.

13.3.2.6 Emergency Communications

Standard

Provisions exist for prompt communications among principal response organizations to emergency personnel and to the public.

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Emergency Plan Evaluation

The Vogtle EP describes the communications systems available for use in emergencies. Communications available in the control room and ERFs include dedicated telephone circuits, normal plant telephone circuits, facsimile line, base station radio console, sound-powered telephones, and the plant page system. This redundancy of communications systems ensures onsite communications during emergencies. Backup power using a battery system is also provided at Vogtle.

The primary means for Vogtle to communicate with GEMA; EPD; Burke, Aiken, Allendale, and Barnwell Counties; and the Savannah River Plant is the ENN, which provides a dedicated ringdown telephone at emergency centers and/or warning points for each of these organizations and agencies. The control room, TSC, EOF, and the backup EOF are all equipped with ENN extensions, and the system is manned at Vogtle and each of these political entities on a 24 hour-per-day basis. The backup communications system for all these facilities is commercial telephone. The applicant's microwave system provides an additional communications link to GEMA. Burke, Aiken, Allendale, and Barnwell Counties as well as EPD and the Savannah River Plant have a remote radio link with Vogtle, which can also be used as a backup communications system.

Primary communication to the NRC Operations Center in Bethesda, Maryland, and NRC Region II in Atlanta, Georgia, is through the ENS, a dedicated NRC telephone system. The Health Physics Network (HPN) commercial direct-line telephone, also provided by NRC, will have extensions in the TSC and the EOF. ENS extensions will be available in the control room, TSC, and EOF. The applicant's microwave system and commercial telephone lines serve as backups to the ENS. Commercial telephone is the primary means of communication between the two hospitals and Vogtle with radio backup through the Burke County Sheriff's Department. The Vogtle ambulance and the Burke County Ambulance Service are equipped with radio communications, and Vogtle can communicate with these vehicles while they are in transit through the hospital radio communications systems. Inplant and offsite monitoring teams communicate with the TSC and EOF via two possible radio frequencies. Inplant communications can also be conducted using inplant telephones.

The communications channels with GEMA; EPD; Burke, Aiken, Allendale, and Barnwell Counties; the Savannah River Plant; and NRC will be tested monthly using the extensions in the control room, TSC, EOF, and backup EOF. In addition, all emergency communications systems will be checked for operability on a quarterly basis. All communications systems and procedures will be tested during an annual communications drill that normally will be conducted during the Vogtle annual emergency exercise.

13.3.2.7 Public Information

Standard

Information is made available to the public on a periodic basis on how they will be notified and what their initial actions should be in an emergency (e.g., listening to a local broadcast station and remaining indoors), the principal points of contact with news media for dissemination of information

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during an emergency (including the physical location or locations) are established in advance, and procedures for coordinated dissemination of information to the public are established.

Emergency Plan Evaluation

The applicant will provide information to the public on the emergency notification and initial actions in the event of an accident at Vogtle. This information will be disseminated at least annually by placing it in local telephone books and posting it in public areas and/or publications distributed by mail within the plume exposure pathway EPZ. The material provided will include information on the NWS tone-alert and siren public notification systems and how they operate; discussion of evacuation and sheltering protective actions; radio stations that will broadcast emergency information; descriptions of evacuation routes and location of reception centers; educational information on radiation; contacts to obtain additional information; and how to meet the special needs of the handicapped. Separate information brochures will be used for Georgia and South Carolina residents. These brochures are essentially the same except for the differences in emergency agencies, telephone numbers, EBS stations, and evacuation routes. These materials will be distributed annually to all residents within the plume exposure pathway EPZ. However, no commitment is made for how or when these materials will be updated. These materials in draft form have been reviewed and appear to provide an adequate and reasonable approach for informing the public about emergencies at Vogtle. However, FEMA has not completed its review of these materials. In addition, the applicant will use the Vogtle Visitor's Center to provide the public education programs on plant operations, plant safety, and radiation.

News centers to coordinate and disseminate information to the media will be established to accommodate representatives from Georgia Power Company, Burke County, the States of Georgia and South Carolina, the Savannah River Plant, and the Federal response agencies. Before the establishment of the emergency news center (ENC) at the Burke County Office Park in Waynesboro, the applicant's public information staff will perform the emergency public information functions from corporate headquarters in Atlanta. As described in Section 13.3.2.2 of this supplement, the staff will relocate at the ENC when the Senior Vice President-Nuclear Operations travels to the site.

A description of the applicant's public information organization is provided in Section G and Appendix 8 of the Vogtle EP. This description provides the identity of the corporate spokesman and other key public information officials; the location of these individuals during the various phases of the emergency; the location of the corporate and site ENCs; the interfaces between the public information staff and other onsite and offsite organizations and authorities; methods for operation of the rumor control program; and the organizational structure of the public information program. It was noted that the alternates for the position of Public Information Manager are different in Section G and Appendix 8 of the EP. This inconsistency was also identified in Section 13.3.2.2 of this supplement.

The applicant will conduct annual training for the public information personnel and will provide media training. The procedures and goals for this training program and a commitment to have the public information staff participate during the annual exercise are provided in the EP.

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The following item requires resolution:

- A commitment should be provided in the EP stating how and when the public information brochures and other materials will be reviewed and updated.

13.3.2.8 Emergency Facilities and Equipment

Standard

Adequate emergency facilities and equipment to support the emergency response are provided and maintained.

Emergency Plan Evaluation

The emergency facilities and equipment are described in Section H of the Vogtle EP. The EP provides for a TSC, an Operational Support Center (OSC), an EOF, and a backup EOF.

The TSC is located outside the control room environmental envelope immediately adjacent to the Unit 1 control room. This facility is to be shared by both units and provides a dedicated center for plant management and technical support personnel to assist the control room personnel during emergencies. The TSC is designed to accommodate 25 individuals including 5 NRC staff members. Figure H-1 of the EP is a simple line drawing of the TSC identifying the main operations area, equipment locations, and work locations of key personnel. The TSC is within a seismic Category I structure and has the same radiological habitability as the control room, although the ventilation system is independent of the control room and is manually operated.

Normal TSC power is provided from offsite normal power through a motor control center backed by the security diesel generator or the battery-backed uninterruptible power system. Battery-operated emergency lighting can provide a minimum of 8 hours of continuous operation.

Two fixed radiation area monitors and portable monitoring equipment and instrumentation are provided for the TSC. Dosimetry, respiratory apparatus, and protective clothing are also available for TSC personnel.

The same document control center is shared between the TSC and the control room. However, separate copies of the Technical Specifications, plant procedures, FSAR, Emergency Plan, and EPIPs are maintained for these facilities.

The OSC is located on the second floor of the maintenance building in the southeast corner, which is normally used as a lunch room. This facility will be used as an assembly area for support and oncoming shift personnel who will assist the control room personnel in an emergency, as well as an assembly point for damage control, repair, and radiological monitoring teams. Figure H-2 of the Vogtle EP is a simple line drawing illustrating the layout of the OSC, the location of the OSC Manager, the briefing area, supply storage, and the OSC conference area.

The emergency teams will obtain their equipment, tools, and supplies from the OSC and either the auxiliary or the maintenance buildings. Drawings, manuals,

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and procedures will be obtained from either the document control room or the maintenance building. Emergency kits with radiation monitoring equipment and instruments, respiratory protective equipment, protective clothing, decontamination supplies, and portable lighting and radio equipment are available. In the event the OSC becomes uninhabitable, the OSC functions will be moved to the TSC, and those personnel not needed for immediate support will be relocated to the EOF.

The EOF is located in the south wing of the Vogtle Training Center, approximately 1.5 miles from the plant. The EOF is the command center for overall management of the emergency and coordination of the radiological assessment of the accident. A simple line drawing of the EOF is provided in Figure H-3 of the EP with 11 rooms labeled as to their function and showing the location of major equipment and key personnel. The EOF can accommodate at least 35 individuals including 1 FEMA and 9 NRC staff members. Space will also be provided for representatives from the Savannah River Plant, the States of Georgia and South Carolina, and Burke County.

The EOF ventilation system is equipped with high efficiency particulate air filters, and the facility can be isolated with recirculating ventilation. The structure has a shielding factor of at least 5.7 for 0.7-MeV gamma radiation.

The normal power supply to the EOF is from the Wilson plant, and backup lighting is provided by battery wall packs that can be operated for 3 hours. The records and information available in the EOF should be adequate for the persons in the EOF to perform their functions. Emergency kits equipped with radiological equipment and instrumentation, respirators, protective clothing, and other supplies are also located in this facility.

The Georgia Power Company district office in Waynesboro, located approximately 16 miles from Vogtle, is designated as the backup EOF if the primary EOF becomes uninhabitable. The transfer of EOF activities will be conducted using a phased approach to ensure that the continuity of dose assessment and decision-making functions is maintained and should take approximately 45 min to complete. Essential data, information, equipment, and instrumentation will be transported with the EOF staff when the relocation occurs.

The control room, TSC, and EOF are equipped with a computerized emergency data acquisition system called the ERF computer system, which includes the safety parameter display system (SPDS) and which has access to the process and effluent radiation monitoring system (PERMS). The PERMS, which provides data on normal and accident conditions at Vogtle to the control room, consists of a minicomputer with data provided by each radiation detector equipped with a data processing module as described in FSAR Section 11.5. The system includes process radiation monitors, effluent radiation monitors, radiation area monitors, airborne radiation monitors, and postaccident radiation monitors. The ERF computer system consists of display cathode ray tubes, line printers, video copiers, a central processing unit located in the power block between the TSC and the control room, and associated hardware and software. This system has the capability to store 2 hours of pre-event data, 12 hours of post-event data, and 2 weeks of additional post-event data with reduced time resolution. The ERF computer system has an unavailability of 0.01, which provides a high degree of reliability.

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Other plant data systems provide meteorological, seismic, and fire detection data to the control rooms. These data are provided to the ERFs by voice communications. Meteorological data include windspeed, wind direction, vertical temperature difference, ambient temperature, dewpoint temperature, and precipitation from a 60-m tower on site. Backup data are obtained from NWS at the Augusta office. Although the agreement letter from DOE indicates that Vogtle and the Savannah River Plant will exchange meteorological data during emergencies, this capability is not discussed in the EP, and no reference is made to the use of these meteorological data as backup material.

The TSC and the OSC will be activated at an Alert or higher classification of emergency, and the EOF will be placed in standby status. The EOF will be activated at the Site Area Emergency or higher emergency classification. All the ERFs have an activation time of about 1 hour.

The contents of the various types of emergency kits for the Vogtle facilities are described in Appendix 4 of the EP. These descriptions include the upper limits of measurement of the instrumentation and types of instrument and equipment provided.

The following item requires resolution:

- A more complete description of the backup systems for obtaining meteorological data should be provided in the EP including the use of data from the Savannah River Plant.

After the above item is resolved, the staff will be able to conclude that the applicant's emergency response facilities and equipment are acceptable on an interim basis for licensing. The staff will confirm the adequacy of the applicant's final ERFs during a postimplementation inspection against the requirements of Supplement 1 of NUREG-0737 on a schedule to be developed between the applicant and the staff.

13.3.2.9 Accident Assessment

Standard

Adequate methods, systems, and equipment for assessing and monitoring actual or potential offsite consequences of a radiological emergency condition are in use.

Emergency Plan Evaluation

Plant systems and effluent parameter values characteristic of a spectrum of off-normal conditions and accidents and the manner in which they are used are provided in Section D of the Vogtle EP and discussed in Section 13.3.2.4 of this supplement. The parameters monitored include reactor cooling system pressure; temperature and pressurizer level; containment pressure and radiation level; and radiation levels in coolant, effluents, and plant areas. Emergency response procedures are provided for quickly assessing core damage, plant systems, and radiation values and for classifying the emergency conditions. Estimates of core damage are derived from the Westinghouse Owner's Group post-accident core damage assessment methodology.

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Inplant radiological measurements using the PERMS and PASS and plant data from the SPDS and other parameters measured on the ERF computer system will be used to assess emergency conditions. The PASS is a Sentry Model B, which has the ability to measure gross gamma radioactivity, gamma spectrum of the radioactivity, boron and chloride concentration, dissolved hydrogen and oxygen, and pH using inline instrumentation and computer-based analysis. The methods used to determine plant conditions are described, and the EPIPs that cover these methods are referenced in the Vogtle EP.

The dose projection system uses a microcomputer that will take meteorological data and release rates from the ERF computer system to obtain whole-body and thyroid doses. The meteorological model used to determine the transport and diffusion of an airborne radioactive release is a variable trajectory Gaussian puff technique. Whole-body and child thyroid doses are used to assess conditions off site. A system of default values is used when release rates or meteorological conditions cannot be determined as described in the EP. The applicant uses its own version of the Interactive Rapid Dose Assessment Model (IRDAM), an NRC-developed computer code, to make these rapid early dose assessments under these conditions with the child thyroid dose as the governing consideration. A manual dose projection method is also provided, and the appropriate EPIP is referenced.

The applicant also uses another computer code (RADDOSE II) for establishing plume trajectories, but this code and the methodology for its use are not described in the EP. Although the EP states that weather forecasts will be obtained from NWS in Augusta, no information on how these forecasts will be used is provided. These dose projection models and systems are available for use in the control room, TSC, and EOF.

Four field monitoring teams, three off site and one on site, can be deployed from the OSC or the EOF and will consist of two individuals each in a Georgia Power Company vehicle equipped with a two-way radio. These vehicles will be maintained on site, and keys will be located in the OSC with keys to the site gasoline supply. The Vogtle monitoring teams will be equipped with gamma and beta/gamma dose rate measuring instruments, respirators, protective clothing, dosimeters, and air sampling equipment listed in Appendix 4 of the EP. The applicant estimates that monitoring teams can be deployed within 1 hour.

Preselected sampling and monitoring locations have been designated around the Vogtle site (as illustrated in Figure iii of the EP) to which these teams will be deployed. Communications will be made at least hourly, and continuous radio communications will be maintained. The fixed thermoluminescent dosimeter (TLD) radiation monitoring and environmental sampling programs are described, and the sampling locations are shown in Figure iii of the EP. However, how the systems in these programs will be used during emergencies is unclear.

Vogtle field monitoring will be coordinated in the EOF with Federal, State, and Savannah River Plant monitoring activities. Laboratory analysis of field samples will be performed for the Vogtle teams either at the site or the Georgia Power Company Smyrna Laboratory with a backup capability available at Hatch. A description of how these field monitoring data will be used to modify dose assessment and estimate ingestion dose is provided.

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The following items require resolution:

- A description of RADDOSE II and its use for establishing plume trajectories should be provided in the EP.
- A description of the use of weather forecasting should be provided in the EP.
- A brief description of how the TLD and environmental sampling systems will be used during emergencies should be provided in the EP.

13.3.2.10 Protective Response

Standard

The range of protective actions has been developed for the plume exposure pathway EPZ for emergency workers and the public. Guidelines for the choice of protective actions during an emergency, consistent with Federal guidance, are developed and in place, and protective actions for the ingestion exposure pathway EPZ appropriate to the locale have been developed.

Emergency Plan Evaluation

All personnel on site at Vogtle will be notified of emergency conditions. Upon notification, personnel with emergency responsibilities will report to their onsite designated stations in the control room and other ERFs. Accountability reports will be provided to the Security Department. Personnel emergency assignment tracking procedures, which include provisions for assignment and reporting logs at the ERFs and the security computer and communications between various facilities, will be used to account for personnel on site.

Nonessential plant personnel, visitors, and contractors will evacuate from the protected area leaving their identification badges at the Security Department. The Security Department will use the computerized accountability system or visual inspection of the badge exchange as well as the accountability reports to determine the accountability of personnel within the protected area within about 30 min. Evacuation of nonessential personnel will be ordered by the Emergency Director when (1) a Site Area or General Emergency is declared, (2) a threat to the health and safety of the personnel is involved, including projected whole-body exposures of 100 mrem or dose rates of 2.5 mrem per hour in nonradiological areas, or (3) there is no reason for the personnel to remain at the site.

Personnel evacuated from the site will travel to an assembly area at the Wilson plant or the Vogtle recreation area in buses and their own vehicles using the evacuation routes shown in Figure ii of the Vogtle EP. Security personnel will check all offices and work locations to ensure that personnel have reported to the assembly areas. If the protected area accountability reveals missing personnel, search and rescue will be initiated to locate these personnel. Procedures for the management and decontamination of personnel and vehicles when they arrive at the assembly areas are described and referenced in the EP. Personnel whose exposure has been detected by portal monitors in the protected area will be decontaminated in the control building. Decontamination waste will be collected, controlled, and returned to the site.

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Potassium iodide will be issued under the direction of a health physics supervisor for projected thyroid exposures of more than 25 rem, and sensitivity to iodide will be determined.

The Vogtle Emergency Director is responsible for ensuring that timely recommendations are provided to offsite officials through initial notifications and followup communications. These recommendations are based on the EPZ protective action guides, and the criteria for airborne releases are provided in Table J-2 of the EP. These criteria are based on the effective dose equivalence of the most critical segment of the population. Protective action recommendations include precautionary sheltering and evacuation of the public and also cover actions for emergency workers.

Evacuation time estimates for Burke County and sheltering factors for various buildings are provided in Tables J-4 and J-3 of the Vogtle EP. Details on the methodology to obtain these estimates as well as evacuation times for the Savannah River Plant and the small areas of South Carolina within the plume exposure pathway EPZ are provided in Appendix 6 of the EP.

A map illustrating the plume exposure pathway EPZ is provided in Figure iii, and a map illustrating the ingestion pathway EPZ is provided in Figure iv of the EP. These maps also give the locations of the plant facilities and the major facilities for offsite emergency support for the applicant, the States of Georgia and South Carolina, and the counties of Burke, Aiken, Barnwell, and Allendale.

13.3.2.11 Radiological Exposure Control

Standard

Means for controlling radiological exposure, in an emergency, are established for emergency workers. The means for controlling radiological exposures shall include exposure guidelines consistent with EPA Emergency Worker and Lifesaving Activity Protective Action Guides.

Emergency Plan Evaluation

Emergency exposure criteria for emergency workers have been established in the event of an accident; they are generally consistent with the EPA Emergency Worker and Lifesaving Activity Protective Action Guides and are provided in Table K-1 of the Vogtle EP. The Vogtle Emergency Director is authorized to permit radiation exposures in excess of 10 CFR 20 limits, but within the limits of Table K-1. Decisions on appropriate exposures and relative risks for the mitigation of accident conditions, damage control, and assessment actions will be made by the Emergency Director in consultation with the health physics staff. The design criteria for equipment and facilities to limit personnel exposure while planned functions, such as PASS sampling, are being performed under emergency conditions are designed with extensive reviews using the as-low-as-reasonably-achievable (ALARA) principle. The design exposure for PASS sampling limits personnel exposure to 60 to 120 mrem for the highest estimated radioactivity level. Where time and urgency allow, the normal radiation work permits will be used. In all cases, a briefing describing the hazards involved in the planned action and the protective actions to be implemented will be given by the

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health physics staff. Logs of the emergency exposures to personnel will be maintained by the emergency health physics supervisor. The methods or procedures for maintaining these exposure records and controlling radiation exposure are described.

All emergency personnel, including offsite personnel responding to the emergency, will be equipped with both TLDs and direct-reading dosimeters. Emergency personnel will be instructed to check their direct-reading dosimeters every 15 to 30 min during the emergency. TLDs will be read monthly or more frequently if the situation warrants, and a 24-hour reading capability is available. The criteria used for more frequent readings of TLDs include an exposure of more than 2.5 rem per quarter as determined by direct-reading dosimeters, loss or offscale indication of direct-reading dosimeters, and the results of radiation surveys.

The Vogtle personnel decontamination facility is located on level 1 of the control building adjacent to the first aid room. The facility is equipped with showers, a gurney, and wash sinks that drain to the plant liquid radwaste system. Personnel decontamination procedures and monitoring techniques for the plant facility, site assembly areas, and offsite medical facilities are described and referenced. The action levels for determining the need for decontamination for both fixed and removable contamination are provided in Table K-2 of the EP. Decontamination procedures for equipment and plant areas are described and referenced in the EP.

Onsite radiological contamination control will be provided by access control by the Security Department to the protected area and by barriers, signs, locked doors, or personnel stationed at the location. The consumption of food, potable liquids, and tobacco will not be permitted in contaminated areas. During emergencies, food and water supplies will be brought on site as required under the direction of the Vogtle Emergency Director.

13.3.2.12 Medical and Public Health Support

Standard

Arrangements are made for medical services for contaminated injured individuals.

Emergency Plan Evaluation

The applicant has made provisions to assist personnel who are injured and are externally contaminated or who may have received high radiation doses. Decontamination materials and first aid kits are available at strategic locations throughout the station. Onshift personnel and onsite and offsite emergency organizations have been trained in the Red Cross Multimedia Course and are available 24 hours per day. The first aid room and the adjacent personnel decontamination facility are in the control building. Personnel will be decontaminated before they are transported to offsite medical facilities unless immediate medical attention is required. Injured personnel will be transported by either the Vogtle ambulance or an ambulance from the Burke County Medical Service as documented in Appendix 2 of the EP.

The applicant has made arrangements with Humana Hospital in Augusta to treat contaminated patients. This hospital has a separate radiation emergency area

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in the burn treatment unit and is equipped for decontamination, exposure evaluation, and contamination control. Burke County Hospital, which also has a separate radiation emergency area and facilities for decontamination and contamination control, is designated as the backup hospital. Agreements with these hospitals and with six local physicians to provide medical treatment are documented in Appendix 2 of the EP. The applicant also has an agreement with the Radiation Management Corporation to provide management of radiation accident victims and consultation services on a 24-hour basis as required and to provide training to the local hospitals and physicians.

A health physics technician will accompany the patient in the ambulance to provide technical support to both the ambulance crew and the hospital staff as described in the EP. This technician is also responsible for controlling and collecting any radioactive waste associated with the injury. If a contaminated patient is transported to a hospital, the vehicle, crew, and contaminated equipment will be returned to Vogtle for decontamination.

13.3.2.13 Recovery and Reentry Planning and Postaccident Operations

Standard

General plans for recovery and reentry are developed.

Emergency Plan Evaluation

The Vogtle EP describes general plans for recovery and reentry operations following any declared emergency condition. The criteria for downgrading from an emergency classification to a recovery condition are:

(1) The reactor is in a stable condition and can be maintained in that condition indefinitely.

(2) The radiation levels are stable or decreasing.

(3) Releases of radioactivity have terminated or are being controlled within permissible limits.

(4) Fires or other conditions no longer constitute a hazard to safety-related systems, equipment, or personnel.

(5) Consultation and coordination with the States of Georgia and South Carolina; Burke, Aiken, Allendale, and Barnwell Counties; NRC; and the Savannah River Plant do not identify any valid reason to remain in an emergency condition.

The transition from an emergency classification to a recovery condition will be declared by the Vogtle Emergency Director. Federal, State, and local officials will be notified that this declaration has been made by means of the emergency communications systems. At that time, the Emergency Director will designate a Recovery Manager to establish the recovery organization. The EP describes by title, the authorities and responsibilities of key personnel in the recovery organization and provides an organizational structure in Figure M-1.

Reentry activities will be controlled in accordance with the criteria in the normal plant procedures, and radiation work practices will follow ALARA program

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principles. TLDs and other dosimetry as appropriate will be issued to all personnel who require access to the plant. These TLDs will be read monthly or more often if work in high radiation areas is undertaken. The criteria for reading TLDs more often are the same as those used during emergencies.

The applicant has developed a system for providing offsite dose assessment and population dose estimates for both the plume exposure and ingestion pathways as described in the appropriately referenced procedures in the Vogtle EP. This system uses plant data on release rates and meteorological dispersion and deposition estimates as well as radiological sampling and monitoring data to estimate population dose in each of the 16 sectors into which the EPZ is divided. This dose information will be provided to offsite authorities to assist them in making their own determinations.

13.3.2.14 Exercises and Drills

Standard

Periodic exercises are (will be) conducted to evaluate major portions of emergency response capabilities, periodic drills are (will be) conducted to develop and maintain key skills, and deficiencies identified as a result of exercises or drills are (will be) corrected.

Emergency Plan Evaluation

The Vogtle EP provides for emergency exercises and drills to evaluate the adequacy of ERFs, equipment, procedures, communications links, emergency personnel response actions, and coordination with offsite emergency response organizations. Some of these drills and exercises will be unannounced. Emergency exercises will test the integrated response and will be conducted in accordance with NRC and FEMA guidance. Exercises will be conducted every 12 months plus or minus 3 months and will be coordinated with Federal, State, and local agencies as appropriate. Exercise scenarios will be varied from year to year to test all major elements of the Vogtle emergency preparedness program within a 5-year period. An exercise will be initiated between 6:00 p.m. and midnight and between midnight and 6:00 a.m. once every 6 years. Although no effort will be made to conduct exercises under various weather conditions, it is expected that they will be conducted under various weather conditions because the date of the exercise will be established several months in advance and may vary by as much as 6 months.

Emergency drills will be conducted to test, develop, and maintain skills in a particular emergency operation. The following types of drills will be conducted:

(1) Communications drills will be conducted by the applicant using the actual message formats to test the understanding of the message content as well as the systems hardware. Contact will be established from the control room, TSC, EOF, and the backup EOF. Drills with all offsite organizations will be conducted annually, normally during the annual emergency exercise, except for the communications drills with Federal organizations and States within the ingestion pathway EPZ, which will be conducted quarterly.

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(2) Fire drills will be conducted in accordance with FSAR Section 13.2.2.1.9 on a quarterly basis. At least one unannounced drill will be conducted annually. Drills will be scheduled to ensure that each member of the fire brigade participates in at least two drills each year.

(3) Medical emergency drills involving a simulated contaminated victim being transported by ambulance to the hospital with subsequent treatment by the hospital staff will be conducted at least annually and may be performed as part of the annual emergency exercise.

(4) Radiological monitoring drills will be conducted annually for onsite and offsite field monitoring. A team will be dispatched to demonstrate proper use of field communications, monitoring equipment, and sampling techniques. The team accompanied by an instructor using a drill scenario will take air samples and calculate the activity as well as collect water, milk, and vegetation samples using the appropriate specific procedures referenced in the EP. The plant communications used to direct the team and report the results also will be tested. This drill normally will be conducted during the annual exercise.

(5) Health physics drills will be conducted semiannually and will involve the response and analysis of simulated airborne and liquid samples as well as direct radiation measurements in the environment. Also, simulated high radiation level liquid sampling using the PASS will be demonstrated. These simulations will be performed with a controller/evacuator using cue cards with radiological data. The specific procedures that will be tested are referenced in the EP. One of these drills may be conducted as part of the annual exercise.

The EP provides for the development of drill and exercise scenarios that include the basic objectives, evaluation criteria, dates, time period, places, participating organizations, simulated events, a time schedule for real and simulated events, a narrative summary, and the assignment of controller/evaluators and Federal observers for exercises. Scenarios for drills will be directed by the corporate Radiological Safety Department or its designee. The scenarios for annual exercises will be prepared under the direction of the Exercise Coordinator and will be coordinated with NRC, FEMA, and other offsite authorities. The results of all drills and exercises will be evaluated, and specific arrangements will be made to allow free play in the decisionmaking process. The Exercise Coordinator or the Radiological Safety Department or its designee is responsible for critiquing exercises and drills, respectively, and reporting results to the Vogtle General Manager to implement corrective actions.

13.3.2.15 Radiological Emergency Response Training

Standard

Radiological emergency response training is provided to those who may be called on to assist in an emergency.

Emergency Plan Evaluation

The Vogtle EP establishes a radiological emergency response training program, which is briefly summarized. The extent of the training program will be

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documented in the Vogtle procedures, and the program follows the Institute of Nuclear Power Operations (INPO) guidelines given in INPO 82-004, "General Employee Training Guideline," dated February 1982. Specialized training will be provided for all personnel who may be called on to respond to an emergency situation. The EP provides for a training program with 13 specific subject areas, which are discussed, for all Vogtle emergency personnel. Table O-2 of the EP provides a matrix of these subject areas and the various emergency positions illustrating which areas constitute required training for the various positions. A review of these training requirements indicates that additional training courses (listed below) should be required for the following positions:

(1) TSC Manager--Core Damage Assessment and Repair and Corrective Actions
(2) Operations Supervisor--Core Damage Assessment
(3) communicators/recorders--Communications and Records
(4) dosimetry team leader--Communications and Records
(5) monitoring team communicator--Communications and Records

The applicant has already committed to include the Communications and Records course for communicators/recorders that was inadvertently omitted from the matrix. Also, there are no training requirements for the TSC engineering staff listed in the requirements matrix.

The EP provides for the training of offsite response groups that may be requested to support fire fighting efforts or provide transportation of injured personnel in the following areas: site orientation, security procedures, exposure evaluation, radiation exposure control, protective clothing and equipment, contamination control, and the nature of radiation. Training also will be offered to State and local emergency response managers with regard to the Vogtle emergency classification system, offsite dose projection, protective action criteria, the Vogtle emergency organization, and the EOF and ENC layout.

Radiological emergency response training will be conducted using lesson plans, classroom lectures, demonstrations of equipment, and walk-throughs. Training courses will be offered throughout the year, and each course will be given at least once each year. The contents of the courses will be tailored to the specific emergency position requirements depending on the group receiving instruction. Drills and exercises are an integral part of the training program and provide for improvements in the training program based on the experience obtained in the previous year's drills. Upon the completion of each training session or drill, participants will be asked to critique the training to ensure continued improvement.

In order for individuals to be qualified to fill a position in the Vogtle emergency organization, they must have received training for the specific subject areas and participated in the appropriate practical drill within the previous 15 months of being assigned to the position.

The following items require resolution:

- The training requirements for the TSC Manager should be expanded to include core damage assessment and repair and corrective actions.

- Core damage assessment should be added to the Operations Supervisor's training requirements.

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- Communications and records training should be added to the requirements of communicators/recorders, dosimetry team leader, and monitoring team communicator.

- The TSC engineering staff should be added to the emergency response training matrix.

13.3.2.16 Responsibility for the Planning Effort: Development, Periodic Review, and Distribution of Emergency Plans

Standard

Responsibilities for plan development and review for distribution of emergency plans are established, and planners are properly trained.

Emergency Plan Evaluation

The Vice President and General Manager-Nuclear Operations has overall responsibility and authority for all nuclear activities including emergency preparedness programs. The corporate manager of radiological safety has the responsibility for emergency planning and preparedness, and the Radiological Safety Department is responsible for developing and maintaining emergency plans and coordinating them with other response organizations. The Emergency Preparedness Coordinator, who reports to the Vogtle Superintendent-Health Physics, coordinates emergency planning and preparedness at Vogtle and in the vicinity of the plant. An organizational chart of the emergency preparedness functions is provided in Figure P-1 of the Vogtle EP.

Approved changes to the Vogtle EP are forwarded to key organizations and appropriate individuals responsible for implementing the EP. Revised pages are dated and marked to indicate where changes have been made, and a description is provided on how these changes are tracked to completion and distributed to ensure that all copies of the EP are updated. Telephone numbers referenced in the EPIPs are updated quarterly, and all EPIPs will be reviewed and certified to be current on an annual basis.

An independent audit of the Vogtle emergency preparedness program, including the EP, EPIPs, emergency preparedness training, annual exercises, drills, equipment, and ERFs, will be conducted annually by the Vogtle Quality Assurance Department under the direction of the Quality Assurance Manager. The Deputy General Manager of Quality Assurance will also direct a review of the interfaces with offsite emergency organizations. Outside contractors with specialized expertise may also be used in conducting these audits. Records of reviews and audits will be maintained in accordance with the Vogtle Technical Specifications.

A list of supporting emergency plans for the Vogtle EP is provided in Section P of the EP. Appendix 9 of the EP provides an index of EPIPs, which appears to be complete except that EPIP 91703-C, "Emergency Support for Hatch Power Plant," is not listed and EPIP 91801 has a title in the index that is different from the title on the document itself.

The following item requires resolution:

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EPIP 91703-C should be added to the index in Appendix 9 of the EP, and the title of EPIP 91801 should be corrected.

13.3.3 Conclusion

On the basis of its review of the Vogtle Emergency Plan to determine conformance with the requirements of Appendix E to 10 CFR 50 and the specific criteria in NUREG-0654/FEMA-REP-1, which addresses each of the planning standards of 10 CFR 50.47(b), the staff concludes that, on the satisfactory correction of those items requiring resolution and commitment as identified in Section 13.3.2 of this supplement, the Vogtle EP will provide an adequate planning basis for an acceptable state of onsite emergency preparedness.

Concerning the matter of offsite medical arrangements for members of the public with respect to the court decision GUARD v. NRC, the applicant will be required to commit to full compliance with the Commission response to the GUARD remand. The staff will report on the resolution of this issue in a supplement to the SER before a license authorizing operation above 5% of rated power is issued.

After a review of the findings and determinations made by FEMA on the adequacy of State and local emergency response plans, and of the revisions to the applicant's EP requiring resolution and commitment as identified in Section 13.3.2 of this supplement, a future supplement to the SER will provide the staff's overall conclusions as to whether the state of onsite and offsite emergency preparedness provides reasonable assurance that adequate protective measures can and will be taken in the event of a radiological emergency.

13.5 Plant Procedures

13.5.2 Operating and Maintenance Procedures

13.5.2.1 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

13.5.2.1.1 Introduction

In the SER, the staff stated that the applicant did not provide descriptions of its emergency operating procedures (EOP) deviations from the generic Westinghouse Owner's Group Emergency Response Guidelines (ERGs). This item was identified as confirmatory item 44. The applicant submitted the EOP deviations in a letter dated September 16, 1985, and provided additional information in its letter dated March 28, 1986. This evaluation describes the staff's review of these submittals.

13.5.2.1.2 Evaluation and Findings

The applicant's deviations from the generic Westinghouse ERGs fell roughly into two categories. The first category includes deviations resulting from plant-specific design differences from the ERG reference plant. These differences did not preclude the use of the Westinghouse ERGs as bases for writing the applicant's EOPs. These differences, however, necessitated some modifications to certain ERG steps when they were converted to EOP steps. The second category

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of deviations includes EOP steps (not included in the ERGs) that were added to enhance the plant-specific emergency procedures. A more detailed discussion of deviations in these two categories follows.

The Vogtle plant has 11 design differences from the high pressure (HP) reference plant on which the Westinghouse ERGs are based. These design differences are discussed in detail in the applicant's submittal of March 28, 1986. Since these features are not included in the reference plant, generic technical guidelines describing the use of these features do not exist. However, the applicant described the guidance that would be given for the use of these particular features. As an example, the applicant's pneumatic power system compressors are not supplied by Class 1E power as in the HP reference plant design. Systems and components that are utilized by the Westinghouse Owner's Group, Revision 1 ERGs, which rely on air, include pressurizer power-operated relief valves (PORVs), steamline PORVs, chemical and volume control system (CVCS) letdown valves, excess letdown valves, and volume control tank makeup control system. Alternative systems and local controls are incorporated into the applicant's plant design to accomplish the functions of these systems without air supply available. The applicant's pressurizer PORVs and steamline PORVs are solenoid-operated valves and electrohydraulically operated valves, respectively, and are powered from Class 1E power sources. Since instrument air is not required to operate the PORVs, some ERG steps had to be rearranged with no significant effect on the final EOPs. The applicant developed alternative methods of charging and letdown using safety-grade equipment and incorporated these methods in the EOPs to be used in the case of loss of instrument air. The staff finds these deviations acceptable because they provide acceptable alternative means for accomplishing the functions required by the ERGs.

Another design difference from the reference plant that necessitated deviations from the ERGs is that the applicant's design does not include a Phase B containment isolation signal. The applicant was asked to identify the components that are isolated on Phase B for the reference plant and the reasons why they are not isolated for Vogtle. The applicant identified the component cooling water (CCW) system as the only system that was isolated on Phase B for the reference plant. For Vogtle, the auxiliary component cooling water (ACCW) system performs the function that the CCW system does for the reference plant. The ACCW system is safety grade and is designed to remain in service during an accident. Therefore, the ERG steps associated with verifying its isolation and restoring flow after the containment isolation signal is cleared are not applicable at Vogtle. The staff finds the deviations from the ERGs related to this design difference acceptable. This plant also differs from the reference plant in the design of the accumulator vent valve arrangement. Vogtle has Class 1E solenoid-operated valves, while the reference plant has air-operated valves. Also, the Vogtle design has two parallel valves on each accumulator and on the common vent line, while the reference plant has a single valve on each accumulator and on the common vent line. The Vogtle design protects against single failure of any valve and provides increased assurance of operation under adverse conditions because it is powered from a Class 1E power supply. The procedure deviations from the ERGs to accommodate this design do not alter the mitigation strategy of the ERGs and are, therefore, acceptable.

The second category of deviations is the incorporation of more detailed instructions for performing certain actions than are provided in the ERGs. For example,

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the Vogtle EOPs give detailed guidance on when to verify that the turbine-driven auxiliary feedwater pump is running (i.e., low-low level in two or more steam generators or a blackout), while the ERGs only say to verify that the turbine-driven pump is running. In addition, the EOPs explain how to start the pump if it is not running. Although a detailed review of all the instructions that constitute this second category of deviations was not performed, the staff finds this level of detail appropriate for plant-specific procedures. On the basis of this finding and the adequacy of the instructions that were reviewed in detail, the staff concludes that the deviations in this category are acceptable.

13.5.2.1.3 Conclusions

The staff has reviewed the applicant's deviations from the Westinghouse Owner's Group Emergency Response Guidelines and finds that these deviations provide (1) acceptable alternative means for accomplishing the accident mitigation strategies described in the ERGs and (2) more detailed instructions that should aid the operator in accomplishing the ERG actions. Therefore, the staff concludes that these deviations are acceptable and considers confirmatory item 44 fully resolved.

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15 ACCIDENT ANALYSES

15.1 Increase in Heat Removal by the Secondary System

15.1.5 System Piping Failures Inside and Outside Containment

15.1.5.1 Steamline Rupture Accident

Reactor Coolant Pump Trip

On June 28, 1985, the staff issued Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,'" to all applicants and licensees with Westinghouse-designed nuclear steam supply systems. The purpose of this generic letter was to inform utilities of the staff's conclusions regarding the Westinghouse Owner's Group (WOG) submittals in response to Generic Letters 83-10c and d and to provide guidance concerning implementation of the reactor coolant pump trip criteria. The WOG reactor coolant pump (RCP) trip methodology allows utilities to select among three alternative RCP trip criteria. The process of criterion selection involves a number of considerations that were assigned plant-specific status by the WOG during the process of the trip criteria review. Accordingly, Generic Letter 85-12 is the mechanism for requesting utilities to select and implement an appropriate RCP trip criterion based on the WOG methodology. This generic letter requests information that was identified in Generic Letters 83-10c and d that has not been provided under the WOG generic review.

The applicant responded to the requests in Generic Letter 85-12 by letter dated February 17, 1986. The staff reviewed this Vogtle-specific submittal and participated in a telephone conference call with the applicant on March 18, 1986. By letter dated April 2, 1986, the staff summarized the conference call and requested additional Vogtle-specific information in order to complete its review. The applicant further responded by letter dated April 21, 1986. The staff has completed a preliminary review of this latest submittal and has concluded that it appears fully responsive to staff questions. The staff does not anticipate that any significant problems will be identified during the remaining review effort. The staff will confirm acceptability of the applicant's response in a detailed review. Pending completion of the staff's detailed assessment, the applicant's response to Generic Letter 85-12 will be identified as a confirmatory item. Resolution will be addressed by the staff in a future supplement to the SER.

15.6 Decreases in Reactor Coolant Inventory

15.6.5 Loss-of-Coolant Accident

15.6.5.1 Radiological Consequences of a Loss-of-Coolant Accident

Post-LOCA Leakage From ESF Systems Outside Containment

In the SER, the staff indicated that the applicant's estimate of 50 gpm for the maximum amount of leakage from emergency core cooling system (ECCS) equipment

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could be substantially reduced. The staff required the applicant to implement a program to reduce ECCS leakage. As noted in the SER, the applicant committed to undertake such a program. The acceptability of the ECCS leakage program was identified as a confirmatory item in the SER.

By letter dated March 21, 1986, the applicant provided a description of the leakage reduction program that included the ECCS. This program will reduce leakage from systems outside containment that might contain highly radioactive fluids to as-low-as-practical levels (NUREG-0737, Item III.D.1.1). The applicant has committed to initiate the program before commercial operation and to implement the program on a continuing basis thereafter.

The staff has reviewed the March 21, 1986, submittal and concludes that although the overall leakage reduction program is still a confirmatory item as discussed in Section 11.5.3 of the SER, it provides the needed assurance that engineered safety features (ESF) leakage, including leakage from ECCS equipment, will be kept as low as practical. This resolves confirmatory item 50. The actual ECCS leak rate data will be reviewed as a part of the leakage program review associated with NUREG-0737, Item III.D.1.1 (confirmatory item 43).

15.8 Anticipated Transients Without Scram

On February 25, 1983, both of the scram circuit breakers at Unit 1 of the Salem Nuclear Power Plant (SNPP) failed to open on an automatic reactor trip signal from the reactor protection system. This incident occurred during plant startup, and the reactor was tripped manually by the operator about 30 sec after the initiation of the automatic trip signal. The failure of the circuit breakers has been determined to be related to the sticking of the undervoltage trip attachment. On February 22, 1983, an automatic trip occurred during startup of SNPP, Unit 1, as the result of steam generator low-low level. In this case, the reactor was tripped manually by the operator almost coincidentally with the automatic trip. On February 28, 1983, the NRC Executive Director for Operations, directed the staff to investigate and report on the generic implications of these occurrences. The results of the staff investigation are reported in NUREG-1000, "Generic Implications of ATWS [Anticipated Transient Without Scram] Events at the Salem Nuclear Power Plant." As a result of this investigation, the Commission requested (by Generic Letter 83-28 dated July 8, 1983) all licensees of operating reactors, applicants for an operating license, and holders of construction permits to respond to certain generic concerns. These concerns are categorized into the following four areas: (1) Post-Trip Review, (2) Equipment Classification and Vendor Interface, (3) Post-Maintenance Testing, and (4) Reactor Trip System Reliability Improvements. The resolution of Generic Letter 83-28 items was identified as an open item in the SER.

Generic Letter 83-28, Item 1.1

The first area, Post-Trip Review, consists of Item 1.1, "Program Description and Procedure," and Item 1.2, "Data and Information Capability." This evaluation addresses Item 1.1 only.

The following review guidelines were developed after initial evaluation of several utility responses to Item 1.1 of Generic Letter 83-28 and incorporate the best features of these submittals. Therefore, these review guidelines effectively represent a "good practices" approach to post-trip review. The staff

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has reviewed the applicant's responses to Item 1.1 against these guidelines:

(1) The licensee or applicant should have systematic safety assessment procedures established that will ensure that the following restart criteria are met before restart is authorized.

- The post-trip review team has determined the root cause and sequence of events resulting in the plant trip.
- Near-term corrective actions have been taken to remedy the cause of the trip.
- The post-trip review team has performed an analysis and determined that the major safety systems responded to the event within specified limits of the primary system parameters.
- The post-trip review has not resulted in the discovery of a potential safety concern (e.g., the root cause of the event occurs with a frequency significantly larger than expected).
- If any of the above re tart criteria are not met, then an independent assessment of the event is performed by the Plant Operations Review Committee (PORC), or another designated group with similar authority and experience.

(2) The responsibilities and authorities of the personnel who will perform the review and analysis should be well defined.

- The post-trip review team leader should be a member of plant management at the shift supervisor level or above and should hold or should have held a senior reactor operator (SRO) license on the plant. The team leader should be charged with overall responsibility for directing the post-trip review, including data gathering and data assessment and he/she should have the necessary authority to obtain all personnel and data needed for the post-trip review.
- A second person on the review team should be a shift technical advisor (STA) or should hold a relevant engineering degree with special transient analysis training.
- The team leader and the STA or engineer should be responsible to concur on a decision/recommendation to restart the plant. A nonconcurrence from either of these persons should be sufficient to prevent restart until the trip has been reviewed by the PORC or equivalent organization.

(3) The licensee or applicant should indicate that the plant response to the trip event will be evaluated and a determination made as to whether the plant response was within acceptable limits. The evaluation should include:

- A verification of the proper operation of plant systems and equipment by comparison of the pertinent data obtained during the post-trip review to the applicable data provided in the FSAR.

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- An analysis of the sequence of events to verify the proper functioning of safety-related and other important equipment. Where possible, comparisons with previous similar events should be made.

(4) The licensee or applicant should have procedures to ensure that all physical evidence necessary for an independent assessment is preserved.

(5) Each licensee or applicant should provide in its submittal, copies of the plant procedures which contain the information required in Items (1) through (4). As a minimum, these should include the following:

- The criteria for determining the acceptability of restart.
- The qualifications, responsibilities, and authorities of key personnel involved in the post-trip review process.
- The methods and criteria for determining whether the plant variables and system responses were within the limits as described in the FSAR.
- The criteria for determining the need for an independent review.

By letters dated March 17, 1986, and April 11, 1986, the applicant provided information regarding its Post-Trip Review Program. The staff has evaluated the applicant's program against the guidelines described above.

With regard to the criteria for determining the acceptability of restart, the applicant will cause each unscheduled reactor trip to be reviewed and classified into one of two categories:

Condition I — The cause of the trip is known and has been corrected. All safety-related equipment functioned properly during the trip.

Condition II — The cause of the trip is not known and/or some safety-related equipment functioned in an abnormal or degraded manner during the trip.

The On-Shift Operations Supervisor (OSOS), who holds an SRO license, and who is the senior management representative for plant operations on shift, has the overall authority for directing the post-trip review process. The STA is responsible for completing a Reactor Trip Report. The OSOS and the STA assess the information for the Reactor Trip Report and concurrently classify the trip as Condition I or Condition II. The OSOS may approve startup following a Condition I trip. For Condition II trips, the Reactor Trip Report is furnished to the Plant Review Board (PRB), which performs an independent assessment of the trip. The Plant General Manager has the authority to authorize restart following a Condition II event and may do so before completion of the independent review by the PRB. The staff concludes that the applicant's criteria for determining the acceptability for restart are in conformance with the guidelines described above and, therefore, are acceptable.

The responsibilities and authorities of the personnel performing the review are well defined. Both the STA and the OSOS must agree on restart following a Condition I trip. The Plant General Manager approves restart following a Condition II trip. The PRB reviews all Condition II trips. The staff finds that

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the applicant's chain of command for responsibility for post-trip review and evaluation is in conformance with the guidelines above and is acceptable.

The applicant has described the information that will be collected and evaluated during the performance of the post-trip review. The information will be analyzed for protection system and overall plant response and will be compared with expected plant response as described in the plant Technical Specifications and FSAR. The staff finds that the applicant's plans for post-trip analysis and evaluation are in conformance with the guidelines above and are acceptable.

As noted above, the applicant has stated that if the cause of the reactor trip is unknown, or if safety-related equipment did not function properly, the trip must be reviewed by the PRB. The applicant has stated that review reports and supporting documentation will be retained for the life of the plant. The staff concludes that the applicant's criteria for independent assessment of unscheduled reactor trips and for preservation of documentation relating to unscheduled trips are in conformance with the guidelines above and are, therefore, acceptable.

Instructions on performing post-trip reviews are contained in plant operating procedure 10006-C, "Reactor Trip Review," which was provided to the staff by letter dated April 11, 1986. The staff has reviewed the procedure and concludes that it is acceptable.

On the basis of its review of the applicant's program description for post-trip review and plant operating procedure 10006-C, "Reactor Trip Review," the staff concludes that the applicant's post-trip review program and procedure are acceptable.

Generic Letter 83-28, Item 4.1

Item 4.1 required licensees and applicants to verify that all vendor-recommended reactor trip breaker modifications have been implemented or to provide a written evalution of the technical reasons for not implementing such modifications.

The only vendor-related modification applicable to the DS-416 reactor trip breakers (RTBs) installed at Vogtle involves the replacement of the undervoltage trip attachment (UVTA) with one of a modified design. By letter dated April 28, 1986, the applicant indicated that the vendor-recommended replacement of the UVTAs has been completed. The staff finds the applicant's response to Item 4.1 acceptable.

Generic Letter 83-28, Item 4.2

Item 4.2 required licensees and applicants to submit a description of their preventive maintenance and surveillance program to ensure reliable reactor trip breaker operation. The description of the submitted program was to include the following:

Item 4.2.1 A planned program of periodic maintenance, including lubrication, housekeeping, and other items recommended by the equipment supplier

Item 4.2.2 Trending of parameters affecting operation and measured during testing to forecast degradation of operation

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The applicant submitted responses to Generic Letter 83-28 on November 8, 1983, and August 1, 1985.

Periodic Maintenance Program

The primary source for periodic maintenance program criteria is Westinghouse Maintenance Program Manual for DS-416 Reactor Trip Circuit Breakers, Revision 0. This document was prepared for the Westinghouse Owner's Group and is the breaker manufacturer's recommended maintenance program for the DS-416 breaker. It provides specific directions with regard to schedules, inspection and testing, cleaning, lubrication, corrective maintenance, and recordkeeping. The document was reviewed to identify those items that contribute to breaker trip reliability consistent with the generic letter. Those items identified for maintenance at 6-month intervals (or when 500 breaker operations have been counted, whichever comes first) that should be included in the applicant's RTB maintenance program are

(1) general inspection to include checking of breaker's cleanliness, all bolts and nuts, pole bases, arc chutes, insulating link, wiring and auxiliary switches

(2) inspection of retaining rings, including those on the UVTA

(3) inspection of arcing and main contacts as specified by the Westinghouse Maintenance Manual

(4) UVTA check as specified by the Westinghouse Maintenance Manual, including replacement of UVTA if dropout voltage is greater than 60% or less than 80% of rated UVTA coil voltage

(5) shunt trip attachment check as specified by the Westinghouse Maintenance Manual

(6) lubrication as specified by the Westinghouse Maintenance Manual

(7) functional check of the breaker's operation before returning it to service

The applicant's RTB periodic maintenance should also include, on a refueling interval basis,

(1) precleaning insulation resistance measurement and recording

(2) RTB dusting and cleaning

(3) postcleaning insulation resistance measurement and recording as specified by the Westinghouse Maintenance Manual

(4) inspection of main and secondary disconnecting contacts, bolt tightness, secondary wiring, mechanical parts, cell switches, instruments, relays, and other panel-mounted devices

(5) UVTA trip force and breaker load check as specified by the Westinghouse Maintenance Manual

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(6) measurement and recording of RTB response time for the undervoltage trip

(7) functional testing of the breaker before returning it to service as specified by the Westinghouse Maintenance Manual

The applicant states that its periodic maintenance program for RTBs includes those items listed above and that all periodic maintenance is performed at those same intervals. The staff finds the applicant's position on Item 4.2.1 acceptable.

Trending of Parameters

Item 4.2.2 of Generic Letter 83-28 specifies that the preventive maintenance and surveillance program is to include trending of parameters affecting operation and measured during testing to forecast degradation of operation. The parameters measured during the maintenance program described above that are applicable for trending are undervoltage trip attachment dropout voltage, trip force, response time for undervoltage trip, and breaker insulation resistance. The staff position is that the above parameters are acceptable, and the staff recommends trending of these parameters to forecast breaker operation degradation or failure. If subsequent experience indicates that any of these parameters are not useful as a tool to anticipate failures or degradation, the licensee or applicant may, with justification and NRC approval, elect not to track that parameter.

The applicant has stated in its August 1, 1985, submittal that trend analysis will be performed on the parameters discussed above. The applicant has indicated that the Maintenance Engineering group will perform trend analysis on an 18-month interval and that maintenance tasks and frequencies will be revised as necessary following evaluation of trending information. The staff finds the applicant's position on Item 4.2.2 acceptable.

Conclusion

On the basis of the above evaluation, the staff concludes that the applicant's responses to Items 1.1, 4.1, 4.2.1, and 4.2.2 are acceptable to partially resolve open item 5.

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17 QUALITY ASSURANCE

17.3 Quality Assurance Program

In the SER, the staff stated that the provisions of regulatory guidance related to quality assurance have been satisfied. However, in Amendment 20 to the FSAR, the applicant revised its commitment to RG 1.94, Revision 1, April 1976, "Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants." This regulatory guide endorses ANSI N45.2.5-1974. Instead, the Vogtle quality assurance program for the design, construction, and operations phases now conforms to the requirements of ANSI/ASME N45.2.5-1978. Because the staff has not endorsed ANSI/ASME N45.2.5-1978, the applicant's commitment to this guide is currently under review by the staff to determine its acceptability for the operations phase. Until this issue is resolved, it is an open item.

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APPENDIX A

CONTINUATION OF CHRONOLOGY OF RADIOLOGICAL REVIEW
OF VOGTLE UNITS 1 AND 2 OPERATING LICENSE REVIEW

October 1, 1985	Atomic Safety and Licensing Board (ASLB) issues Memorandum and Order (Ruling on Applicants' Motion of September 5, 1985, for Reconsideration and Clarification).
October 2, 1985	Letter from applicant concerning SER open item 10, "Cathodic protection of fuel oil tanks."
October 3, 1985	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Contention 8 re: Vogtle Quality Assurance).
October 4, 1985	Letter to applicant requesting additional information on SER confirmatory item 14f, "Dependability of containment isolation (purge valves)."
October 7, 1985	Letter from applicant concerning anticipated transients without scram.
October 10, 1985	Letter to applicant requesting additional information on SER confirmatory item 5, "Final details of groundwater and settlement monitoring program."
October 15, 1985	Letter to applicant concerning second lined startup pond.
October 15, 1985	Letter from applicant concerning witnessing of piping system pressure tests.
October 16-17, 1985	Meeting with applicant to discuss staff comments on the Emergency Plan (SER open item 13). (Summary issued October 28, 1985.)
October 18, 1985	ASLB issues Order granting extension of time to file a request for reconsideration of the Memorandum and Order (Ruling on Motion for Summary Disposition of Contention 8 re: Vogtle Quality Assurance).
October 21, 1985	Meeting with applicant to provide guidance regarding the Technical Specifications. (Summary issued October 28, 1985.)
October 21, 1985	Letter from applicant providing emergency public information/education materials.
October 22, 1985	Letter to applicant transmitting report and requesting additional information on response to SER open item 5, "Generic Letter 83-28," Items 1.1 and 1.2.

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October 22, 1985	Letter to applicant requesting additional information related to the Emergency Plan (SER open item 13).
November 5, 1985	Letter to applicant forwarding Standard Technical Specifications.
November 5, 1985	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Contention 10.7 re: Hydrogen Recombiners).
November 11, 1985	Letter from applicant regarding SER open item 7a, "Fire doors and dampers."
November 12, 1985	Letter from applicant concerning SER confirmatory item 1, "Correlation and analysis of data from old and new meteorological towers."
November 12, 1985	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Contention 7 re: Groundwater Contamination).
November 14, 1985	Letter to applicant forwarding Supplement 1 to the SER.
November 15, 1985	Letter from applicant concerning SER confimatory item 5, "Details of groundwater and settlement monitoring program."
November 15, 1985	Letter from applicant concerning SER open item 13, "Emergency preparedness."
November 18, 1985	Letter to applicant concerning use of American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) Case N-412.
November 18, 1985	Letter from applicant concerning spectral broadening procedures of ASME Code Case N-397.
November 22, 1985	Letter from applicant concerning Generic Letter 81-01.
November 25, 1985	ASLB issues Memorandum and Order (Ruling Upon Motion for Summary Disposition of Contention 14 re: TDI Emergency Diesel Generators).
November 26, 1985	Letter to applicant requesting additional information on safety parameter display system (SER open item 14).
November 26, 1985	Letter from applicant concerning SER open item 4, "Toxic gas evaluation of chemicals."
December 3, 1985	Generic Letter 85-22--Potential for Loss of Post-LOCA Recirculation Capability Due to Insulation Debris Blockage.
December 3, 1985	ASLB issues Memorandum and Order (Ruling on Joint Intervenors' Motion for Reconsideration and Other Relief).
December 9, 1985	Letter from applicant regarding the fire protection program.

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December 9, 1985	Letter from applicant concerning SER open item 14a, "Detailed control room design review."
December 9, 1985	Letter from applicant concerning SER confirmatory item 22, "Operator action in event of a small-break LOCA."
December 11, 1985	Meeting with applicant to discuss emergency action levels (SER open item 13). (Summary issued December 17, 1985.)
December 18, 1985	Letter from applicant forwarding FSAR Amendment 20.
December 24, 1985	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Joint Intervenors' Contention 12 re: Cooling Tower Drift).
December 26, 1985	Letter to applicant accepting use of ASME Code Case N-397.
December 31, 1985	Letter from applicant forwarding Revision 3 to the Emergency Plan.
January 2, 1986	Letter from applicant concerning SER open item 13, "Emergency preparedness."
January 6, 1986	ASLB issues Memorandum and Order (Ruling on Motion for Partial Reconsideration of Memorandum and Order Ruling on Motion for Summary Disposition of Contention 7).
January 6, 1986	Letter from applicant concerning SER open item 14a, "Detailed control room design review."
January 7, 1986	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Contention 10.5 re: ASCO Solenoid Valves).
January 8, 1986	ASLB issues Order Compelling Discovery.
January 8, 1986	Letter from applicant forwarding corrected version of FSAR Amendment 20.
January 14, 1986	ASLB issues Scheduling Order. Evidentiary hearing will commence on March 11, 1986, in Waynesboro, Georgia.
January 16, 1986	Letter from applicant concerning SER open item 7a, "Fire doors and dampers."
January 17, 1986	Letter from applicant concerning SER open item 4, "Toxic gas evaluation of chemicals."
January 21, 1986	Letter from applicant concerning SER open item 6, "Emergency response capability--RG 1.97, Rev. 2."
January 23, 1986	ASLB issues Supplemental Scheduling Order. January 14, 1986, Scheduling Order is made applicable to Contention 10.1 just as it is to Contentions 7 and 10.5.

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January 23, 1986	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Contention 10.1 re: Dose Rate Effects).
January 29, 1986	ASLB issues Notice of Hearing. Hearing will commence on March 11, 1986, at 9:30 a.m., in Waynesboro, Georgia.
January 29-30, 1986	Electrical, Instrumentation, and Control Systems Branch site visit. (Summary issued February 13, 1986.)
February 3, 1986	ASLB issues Memorandum and Order (Ruling on Applicants' Motion for Summary Disposition of Contention EP-6.)
February 6, 1986	Letter from applicant concerning SER confirmatory item 5, "Details of groundwater and settlement monitoring program."
February 7, 1986	Letter from applicant concerning automatic sprinkler system inside containment and safe shutdown analysis.
February 13, 1986	ASLB issues Memorandum and Order (Ruling on Applicants' Motion for Partial Reconsideration of Board Ruling re: Contention 10.1)
February 14, 1986	Letter from applicant responding to emergency planning items.
February 14, 1986	Letter from applicant forwarding Revision 4 to Emergency Plan.
February 17, 1986	Letter from applicant forwarding response to Generic Letter 85-12.
February 18, 1986	Letter from applicant forwarding State and local emergency response plans.
February 18, 1986	Letter from applicant forwarding Amendment 21 to FSAR.
February 19, 1986	Letter from applicant forwarding Topical Report WCAP-10529, "Cold Overpressure Mitigating System."
February 19, 1986	Letter from applicant concerning SER open item 4, "Toxic gas evaluation of chemicals."
February 20, 1986	Letter to applicant requesting additional information on SER confirmatory item 44, "Procedures generation package."
February 21, 1986	Letter from applicant concerning SER confirmatory item 39, "Process control program."
February 25, 1986	Letter from applicant forwarding Revision 5 to Emergency Plan.
February 26, 1986	Meeting with applicant to discuss safety parameter display system and detailed control room design review (SER open item 14). (Summary issued March 10, 1986.)

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February 26, 1986	Letter from applicant forwarding Amendment 22 to FSAR.
February 27, 1986	Letter from applicant concerning RG 9.3 information.
February 28, 1986	Letter from applicant concerning errata to Geotechnical Verification Work--Report of Results.
February 28, 1986	Board Notification 86-08--Summary Board Notification for Vogtle Units 1 and 2.
February 28, 1986	Letter from applicant forwarding draft Technical Specifications.
March 5, 1986	Letter from applicant concerning SER confirmatory item 15, "Compliance with NUREG-0737, Item II.F.2."
March 5, 1986	Letter to applicant concerning reports on probabilistic risk assessment insights.
March 5, 1986	ASLB issues Prehearing Conference Order.
March 6, 1986	ASLB issues Memorandum and Order (Ruling upon Applicants' Motion of February 10, 1986 for Summary Disposition of Contention EP-7 re: Emergency Response Plans).
March 17, 1986	Letter from applicant concerning SER open item 5, "Generic Letter 83-28."
March 17, 1986	Letter from applicant forwarding suggested SER corrections.
March 21, 1986	ASLB issues Order (Granting NRC Staff's Motion for Extension of Time To Respond to Applicants' Motion for Summary Disposition).
March 21, 1986	Letter from applicant concerning SER confirmatory items 43, "Compliance with NUREG-0737, Item III.D.1.1," and 50, "Program to minimize ECCS equipment leakage."
March 26, 1986	Letter from applicant concerning pump and valve operability assurance.
March 26, 1986	Letter from applicant concerning SER open item 5, "Generic Letter 83-28."
March 26, 1986	Letter from applicant forwarding correction to Revision 5 of Emergency Plan.
March 26, 1986	Letter from applicant concerning modifications to the power block backfill.
March 28, 1986	Letter from applicant concerning SER confirmatory item 44, "Procedures generation package."

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April 2, 1986	Letter to applicant requesting additional information on Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5."
April 3, 1986	Letter from applicant concerning SER confirmatory item 37, "Procedures for ordering fuel after 5 days."
April 4, 1986	Letter from applicant concerning training for licensed operator and nonlicensed plant staff.
April 4, 1986	Letter from applicant concerning SER confirmatory item 6, "Verification of FSAR commitments on Category 1 backfill."
April 4, 1986	ASLB issues Memorandum and Order (Ruling on Motion for Summary Disposition of Intervenors' Contention EP-2/EP-2(c) (Use of Tone Alert Radios))
April 8, 1986	Meeting with applicant to discuss fire damper deviations. (Summary issued April 18, 1986.)
April 9, 1986	Letter to applicant transmitting first draft of Technical Specifications.
April 11, 1986	Letter from applicant regarding Item 1.1 of Generic Letter 83-28 (SER open item 5).
April 15, 1986	Letter from applicant forwarding Seismic Qualification Report in response to SER open item 1, "Equipment qualification."
April 16, 1986	Letter from applicant concerning code addenda of record for piping stress analysis.
April 17, 1986	Letter from applicant forwarding Amendment 23 to FSAR.
April 18, 1986	Letter from applicant forwarding Annual Financial Reports.
April 21, 1986	Letter from applicant forwarding response to Generic Letter 85-12.
April 21, 1986	Letter from applicant on shift turnover procedures in response to SER open item 11, "Licensee qualifications for operation."
April 21, 1986	Letter from applicant on SER confirmatory item 35, "Procedures for load following test and/or maintenance."
April 22, 1986	ASLB issues Order (transcript corrections).
April 22, 1986	Letter from applicant submitting emergency public information/education materials.
April 24, 1986	Letter to applicant requesting additional information on SER confirmatory items 39 and 43.

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April 24, 1986	Letter from applicant listing major topics pertaining to the proposed Technical Specifications.
April 25, 1986	Letter from applicant regarding SER open item 6, "Emergency response capability--RG 1.97, Rev. 2."
April 25, 1986	ASLB issues Order Granting NRC Staff's Request for Postponement in Regard to Applicants' Motion for Summary Disposition of Contention EP-5.
April 28, 1986	Letter from applicant concerning SER open item 5, "Generic Letter 83-28."
April 29, 1986	ASLB issues Memorandum and Order (Ruling on Applicants' Motion for Summary Disposition of Contention EP-4).

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APPENDIX B

REFERENCES

Code of Federal Regulations, Title 10, "Energy," U.S. Government Printing Office, Washington, D.C. (contains General Design Criteria).

Dutton, D., Georgia Power Company, letter to E. G. Adensam, NRC, "Supplement 1 to NUREG-0737 (Generic Letter 82-33)," April 14, 1983.

Grimes, B., NRC, letter to all power reactor licensees, "Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978.

Institute of Nuclear Power Operations, INPO 82-004, "General Employee Training Guideline," February 1982.

Rahe, E. P., Westinghouse, letter to H. R. Denton, NRC, "Topical Reports WCAP-8822-P-S2 and WCAP-8860-S2, 'Mass and Energy Releases Following a Steamline Rupture,'" October 7, 1985.

---, letter to H. R. Denton, NRC, "Additional Information on Topical Reports WCAP-8822-P-S2 and WCAP-8860-S2, 'Impact of Steam Superheat in Mass/Energy Releases Following a Steamline Rupture for Dry and Subatmospheric Containment Designs,'" December 13, 1985.

Rossi, C. E., NRC, letter to E. P. Rahe, Westinghouse, "Acceptance for Referencing of Licensing Topical Reports WCAP-8822-P-S1/WCAP-8860-S1 and WCAP-8822-P-S2/WCAP-8860-S2, 'Mass and Energy Release Following a Steam Line Rupture,'" May 27, 1986.

U.S. Nuclear Regulatory Commission, 47 FR 30232, "Emergency Planning and Preparedness, Final Rule," July 13, 1982.

---, Generic Letter 82-33, "Supplement 1 to NUREG-0737--Requirements for Emergency Response Capability," December 17, 1982.

---, Generic Letters 83-10c and d, "Resolution of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,'" February 8, 1983.

---, Generic Letter 83-28, "Required Actions Based on Generic Implications of Salem ATWS Events," July 8, 1983.

---, Generic Letter 85-12, "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,'" June 28, 1985.

---, NUREG-0654/FEMA-REP-1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," Revision 1, November 1980.

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---, NUREG-0737, "Clarification of TMI Action Plan Requirements," November 1980; Supplement 1, January 1983 (also issued as Generic Letter 82-33, December 17, 1982).

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition," July 1981 (contains Branch Technical Positions).

---, NUREG-1000, "Generic Implications of ATWS Events at the Salem Nuclear Power Plant," April 1983.

---, Office of Inspection and Enforcement, IE Bulletin 79-15, "Deep Draft Pump Deficiencies," July 11, 1979.

---, IE Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls," March 13, 1980.

---, IE Inspection Report 50-424/85-35, December 18, 1985.

---, IE Inspection Report 50-424/85-51, December 13, 1985.

---, IE Inspection Report 50-424/86-12 and 50-425/86-18, May 9, 1986.

Westinghouse Electric Corporation, WCAP-8822-S2 (WCAP-8860-S2, nonproprietary version), J. C. Butler and P. A. Linn, "Mass and Energy Release Following a Steamline Rupture," September 1985.

---, WCAP-10529, R. Fleming, "Cold Overpressure Mitigating System," February 1984.

INDUSTRY CODES AND STANDARDS

American National Standards Institute, ANSI N45.2.5-1974, "Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants."

American National Standards Institute/American Nuclear Society, ANSI N18.1-1971 (ANSI/ANS N18.1-1971), "Selection and Training of Nuclear Power Plant Personnel."

American National Standards Institute/American Society of Mechanical Engineers, ANSI/ASME N45.2.5-1978, "Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete, Structural Steel, Soils, and Foundations During the Construction Phase of Nuclear Power Plants."

American Society for Testing and Materials, ASTM D2113-1983, "Standard Practice for Diamond Core Drilling for Site Investigation."

National Fire Protection Association (NFPA) 13, "Standard for the Installation of Sprinkler Systems," 1983.

Underwriters Laboratory, Standard 555, "Standard for Fire Dampers and Ceiling Dampers," 1979.

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APPENDIX D

ACRONYMS AND INITIALISMS

ACCW	auxiliary component cooling water
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
ASLB	Atomic Safety and Licensing Board
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transients without scram
BRH	Bureau of Radiological Health
BTP	branch technical position
CCW	component cooling water
CFR	Code of Federal Regulations
COMS	cold overpressure mitigation system
CVCS	chemical and volume control system
DG	diesel generator
DNB	departure from nucleate boiling
DNR	Department of Natural Resources
DOE	Department of Energy
EAL	emergency action level
EBS	emergency broadcast system
ECCS	emergency core cooling system
ENC	emergency news center
ENN	Emergency Notification Network
ENS	emergency notification system
EOC	Emergency Operations Center
EOF	Emergency Operations Facility
EOP	emergency operating procedure
EP	Emergency Plan
EPA	Environmental Protection Agency
EPD	Emergency Preparedness Division
EPIA	Emergency Preparedness Implementation Appraisal
EPIP	emergency preparedness implementing procedure
EPZ	emergency planning zone
ERF	Emergency Response Facility
ERG	Emergency Response Guideline
ESF	engineered safety feature(s)
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
GDC	general design criterion(a)
GEMA	Georgia Emergency Management Agency

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HP	high pressure
HPN	Health Physics Network
HVAC	heating, ventilation, and air conditioning
IE	Office of Inspection and Enforcement
INPO	Institute of Nuclear Power Operations
IRDAM	Interactive Rapid Dose Assessment Model
LOCA	loss-of-coolant accident
MSLB	main steamline break
NFPA	National Fire Protection Association
NRC	U.S. Nuclear Regulatory Commission
NSCW	nuclear service cooling water
NUE	Notification of an Unusual Event
NWS	National Weather Service
OSC	Operational Support Center
OSOS	On-Shift Operations Supervisor
PASS	postaccident sampling system
PERMS	process and effluent radiation monitoring system
PORC	Plant Operations Review Committee
PORV	power-operated relief valve
PRB	Plant Review Board
RCP	reactor coolant pump
RG	regulatory guide
RHR	residual heat removal
RTB	reactor trip breaker
RVHVS	reactor vessel head vent system
SER	Safety Evaluation Report
SGTR	steam generator tube rupture
SNPP	Salem Nuclear Power Plant
SPDS	safety parameter display system
SRO	senior reactor operator
SRP	Standard Review Plan
SSER	supplement to Safety Evaluation Report
STA	shift technical advisor
TDI	Transamerica Delaval, Inc.
TER	Technical Evaluation Report
TLD	thermoluminescent dosimeter
TMI	Three Mile Island
TSC	Technical Support Center
UHS	ultimate heat sink
UL	Underwriters Laboratory
UVTA	undervoltage trip attachment
VRS	volume reduction system
WOG	Westinghouse Owner's Group

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APPENDIX E

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APPENDIX K

ERRATA TO VOGTLE SAFETY EVALUATION REPORT
AND ITS SUPPLEMENT 1

Page	Paragraph*	Line	Change
Safety Evaluation Report			
2-15	1	3	Change "13" to "16"
2-18	6	2	Change "24" to "14"
3-3	6	5	Change "BC-TOP-3" to "BC-TOP-3A"
3-4	2	1	Change "BC-TOP-3" to "BC-TOP-3A"
3-5	3	1	Change "BC-TOP-3" to "BC-TOP-3A"
3-31	2	2	Delete "and constructed"
4-21	1	6	Change "fraction" to "percentage"
5-13	3	7	Change "gaseous" to "liquid"
5-28	2	4	Change "flow to" to "flow from"
5-32	1	2	Change "gaseous" to "liquid"
9-8	4	2	Change "spent fuel pool bridge" to "fuel handling"
9-12	2	6	Delete "and"
9-42	4	3	Add "banks of" after "two"
9-63	2, Item (2)	3	Change "below" to "before; delete "where"
9-63	2, Item (2)	4	Change "roof" to "wall"
9-64	1, Item (2)	1	Change "on the roof" to "immediately outside the wall"
11-9	1	1	Delete "regenerative"
13-19	5	8	Insert "substantially" after "be"

*Full paragraph

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Page	Paragraph*	Line	Change
Supplement 1			
2-1	3	14	Insert "clean backfill sands with less than 6% passing the No. 200 sieve and 3.5 lb/ft^3 for the" before "backfill"

*Full paragraph

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APPENDIX M

CONFORMANCE TO REGULATORY GUIDE 1.97
VOGTLE ELECTRIC GENERATING PLANT, UNIT NOS. 1 AND 2

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EGG-EA-6806

CONFORMANCE TO REGULATORY GUIDE 1.97
VOGTLE ELECTRIC GENERATING PLANT, UNIT NOS. 1 AND 2

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ABSTRACT

This EG&G Idaho, Inc., report reviews the submittals for Regulatory Guide 1.97 for Unit Nos. 1 and 2 of the Vogtle Electric Generating Plant and identifies areas of nonconformance to the regulatory guide. Exceptions to Regulatory Guide 1.97 are evaluated.

Docket Nos. 50-424 and 50-425

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FOREWORD

This report is supplied as part of the "Program for Evaluating Licensee/Applicant Conformance to RG 1.97," being conducted for the U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Division of PWR Licensing-A, by EG&G Idaho, Inc., NRR and I&E Support Branch.

The U.S. Nuclear Regulatory Commission funded the work under authorization B&R 20-19-40-41-3.

Docket Nos. 50-424 and 50-425

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CONFORMANCE TO REGULATORY GUIDE 1.97
VOGTLE ELECTRIC GENERATING PLANT, UNIT NOS. 1 AND 2

1. INTRODUCTION

On December 17, 1982, Generic Letter No. 82-33 (Reference 1) was issued by D. G. Eisenhut, Director of the Division of Licensing, Nuclear Reactor Regulation, to all licensees of operating reactors, applicants for operating licenses and holders of construction permits. This letter included additional clarification regarding Regulatory Guide 1.97, Revision 2 (Reference 2), relating to the requirements for emergency response capability. These requirements have been published as Supplement No. 1 to NUREG-0737, "TMI Action Plan Requirements" (Reference 3).

Georgia Power Company, the applicant for the Vogtle Electric Generating Plant (VEGP), provided a response to the generic letter on April 14, 1983 (Reference 4). This submittal refers to the Vogtle Final Safety Analysis Report (FSAR, Reference 5) for a review of the instrumentation provided for conformance to Regulatory Guide 1.97. On March 15, 1985 (Reference 6), on June 20, 1985, (Reference 7), on September 6, 1985 (Reference 8) and on January 21, 1986 (Reference 9), the applicant provided additional information.

This report provides an evaluation of these submittals.

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2. REVIEW REQUIREMENTS

Section 6.2 of NUREG-0737, Supplement No. 1, sets forth the documentation to be submitted in a report to the NRC describing how the applicant complies with Regulatory Guide 1.97 as applied to emergency response facilities. The submittal should include documentation that provides the following information for each variable shown in the applicable table of Regulatory Guide 1.97.

1. Instrument range
2. Environmental qualification
3. Seismic qualification
4. Quality assurance
5. Redundancy and sensor location
6. Power supply
7. Location of display
8. Schedule of installation or upgrade

The submittal should identify deviations from the regulatory guide and provide supporting justification or alternatives.

Subsequent to the issuance of the generic letter, the NRC held regional meetings in February and March 1983 to answer licensee and applicant questions and concerns regarding the NRC policy on this subject. At these meetings, it was noted that the NRC review would only address exceptions taken to Regulatory Guide 1.97. Where licensees or applicants explicitly state that instrument systems conform to the regulatory guide it was noted that no further staff review would be necessary. Therefore, this

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report only addresses exceptions to Regulatory Guide 1.97. The following evaluation is an audit of the applicant's submittals based on the review policy described in the NRC regional meetings.

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3. EVALUATION

In the applicant's response to NRC generic letter 82-33, Section 7.5 of the FSAR is identified as containing (a) the description of the post-accident monitoring system (PAMS), (b) tables that identify the monitored parameters, and (c) identification of compliance to or deviations from Regulatory Guide 1.97 along with the supporting justification or alternatives. This evaluation is based on the information provided in Section 7.5 of the FSAR, and the March 15, 1985, the June 20, 1985, the September 6, 1985 and the January 21, 1986 submittals.

3.1 Adherence to Regulatory Guide 1.97

Within Table 7.5.2-1 of the FSAR, the applicant has identified where the post-accident monitoring instrumentation conforms to Regulatory Guide 1.97, Revision 2, and where deviations exist. Therefore, we conclude that the applicant has provided an explicit commitment on conformance to Regulatory Guide 1.97 prior to startup. Exceptions to and deviations from the regulatory guide are noted in Section 3.3.

3.2 Type A Variables

Regulatory Guide 1.97 does not specifically identify Type A variables, i.e., those variables that provide the information required to permit the control room operator to take specific manually controlled safety actions. The applicant classifies the following instrumentation as Type A:

1. Reactor coolant system pressure (wide range)

2. Reactor coolant system wide range hot leg temperature

3. Reactor coolant system wide range cold leg temperature

4. Wide range steam generator water level

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5. Narrow range steam generator water level

6. Pressurizer level

7. Containment pressure

8. Steamline pressure

9. Refueling water storage tank level

10. Containment water level--narrow range

11. Containment water level--wide range

12. Condensate storage tank level

13. Auxiliary feedwater flow

14. Containment radiation level--wide range

15. Containment radiation level--narrow range

16. Steamline radiation monitor

17. Core exit temperature

18. Degrees of subcooling

These variables meet the Category 1 recommendations consistent with the requirements for Type A variables.

3.3 Exceptions to Regulatory Guide 1.97

The applicant identified deviations and exceptions from Regulatory Guide 1.97. These are discussed in the following paragraphs.

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3.3.1 Reactor Vessel Water Level

In the initial submittal, the applicant took exception to Regulatory Guide 1.97 for the reactor vessel water level variable. Category 2 instrumentation was indicated instead of the recommended Category 1 instrumentation.

In the June 20, 1985 submittal, the applicant states that Table 7.5.2-1 of the FSAR will be revised in Amendment 17 to reflect that the reactor vessel water level system conforms to Regulatory Guide 1.97, Rev. 2, i.e., meeting Category 1 criteria. We conclude that the clarification provided by the applicant in the March 15, 1985 and June 20, 1985 submittals shows that this instrumentation is acceptable.

3.3.2 Containment Isolation Valve Status

In the initial submittal, the applicant indicates that the instrumentation for this variable is Category 2 instead of the recommended Category 1 instrumentation.

The applicant provided additional information within the March 15, 1985 submittal stating that containment isolation valve status conforms to Regulatory Guide 1.97, Rev. 2, Category 1 instrumentation criteria, except for a single indication per valve. From the information provided, we find that the applicant deviates from a strict interpretation of the Category 1 redundancy recommendation. Only the active valves have position indication (i.e., check valves have no position indication). Since redundant isolation valves are provided, we find that redundant indication per valve is not intended by the regulatory guide. Position indication of check valves is specifically excluded by Table 2 of Regulatory Guide 1.97. Therefore, we find that the instrumentation for this variable is acceptable.

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3.3.3 Pressurizer Heater Status

Within the initial submittal, the applicant does not identify instrumentation for monitoring current to the pressurizer heaters. The applicant states that indication of the pressurizer heater breaker position is adequate indication to the operator that the pressurizer heaters are operable.

In Reference 9, the applicant commits to provide the recommended current monitoring instrumentation to those heater banks that are capable of being supplied emergency power from the diesel generators. We find this instrumentation acceptable.

3.3.4 Accumulator Tank Level and Pressure

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable with ranges of 10 to 90 percent volume and 0 to 750 psig. The applicant has identified Category 3 instrumentation for level with a range of 59 to 66 percent volume, and Category 2 instrumentation for pressure with a range of 0 to 700 psig.

The applicant has identified the accumulator pressure as the key variable to monitor the operation of the accumulators. The range is justified because two check valves in series on the discharge line prevents pressures from exceeding the manually controlled pressure. The nitrogen cover gas is manually kept between 564 and 637 psig.

As a backup variable, the category and the range of the level instrumentation is adequate, as the regulatory guide accepts Category 3 backup variables.

The accumulators are passive devices. Their discharge into the RCS is actuated solely by a decrease in RCS pressure. We find that the

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instrumentation supplied for this variable is adequate to determine that the accumulators have discharged. Therefore, the ranges of this instrumentation are acceptable for this variable.

3.3.5 Containment Atmosphere Temperature

Within the initial submittal, the applicant specifies Category 3 instrumentation instead of Category 2. The applicant states that this is based on the fact that the plant emergency response guidelines do not require the operator to take action that would result in adverse consequences if the containment temperature were indicating an erroneous value.

The applicant's justification was expanded in the March 15, 1985 submittal. The applicant states that the key variables for monitoring containment cooling are containment spray flow, containment water level, containment spray valve status, containment fan cooler damper position, containment fan cooler breaker position, and containment pressure. Because the operator will not use the containment atmosphere temperature as a key variable, and the applicant has identified alternate key variables to monitor containment cooling, we find the deviation from Category 2 to Category 3 for this variable acceptable.

3.3.6 Containment Sump Water Temperature

Regulatory Guide 1.97 recommends Category 2 instrumentation for this variable. The applicant has instrumentation for this variable. The sump temperature instruments are Category 2 except for the electrical connectors inside containment that are not qualified for a steam environment. The residual heat removal (RHR) heat exchanger inlet temperature (essentially the same temperature as the sump) instruments are Category 2 except for high radiation levels. The applicant states that further qualification of

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these instruments will not increase the safety of the station or increase the ability to mitigate the consequences of an accident or to bring the unit to a safe shutdown.

The applicant points out the following justifications.

1. There are no operator actions that are dependent on sump temperature.

2. Recirculation flow is initiated based on the refueling water storage tank level.

3. The RHR system is the only system that can reduce the sump water temperature. The decision to use this system is based on reactor coolant system (RCS) temperature, inadequate cooldown rate or RCS pressure.

4. The RHR pumps will have adequate net positive suction head regardless of the sump temperature.

5. Should a quantitative measure of heat removal be desired, the supplied instrumentation can be used.

The applicant does not have instrumentation or alternate instrumentation for this variable that is fully qualified to the Category 2 requirements. Thus, in a post-accident situation, a quantitative measure of the heat removal by way of the containment sump would not necessarily be available. Because of this, the applicant's justification is not acceptable.

The applicant should therefore provide instrumentation for this variable that is environmentally qualified in accordance with the provisions of 10 CFR 50.49 and Regulatory Guide 1.97.

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3.3.7 Heat Removal by the Containment Fan Heat Removal System

The applicant indicates that this variable is not required. The applicant states that (a) the containment spray flow indication, (b) the containment spray system valve status indication, (c) the containment pressure indication, (d) the containment water level indication, (e) the containment spray pump status indication, (f) the containment fan cooler damper position indication, and (g) the containment fan cooler breaker position, all provide indication to allow the operator to determine operability. These variables are identified by the applicant as Category 2. Based on the above diversity, we find the alternate instrumentation acceptable.

3.3.8 Accident Sampling (Primary Coolant, Containment Air and Sump)

Within the initial submittal the applicant notes an exception for this variable. The applicant did not provide the information required by Section 6.2 of Supplement No. 1 to NUREG-0737.

In the applicant's June 20, 1985 submittal, the information required by Section 6.2 of Supplement No. 1 to NUREG-0737 is provided. The applicant states that this information will be included in Table 7.5.2-1 of the FSAR by Amendment 17. Review of the information provided by the applicant on the accident sampling capability indicates that the applicant conforms with Regulatory Guide 1.97 for this variable.

3.3.9 Boric Acid Charging Flow

The applicant indicates that this variable is not required. The justification is that (a) the refueling water storage tank level indication, (b) the high-head safety injection flow indication, (c) the low-head safety injection flow indication, (d) the containment water level indication, and (e) the emergency core cooling system (ECCS) valve status, monitor the performance of the ECCS. The normal charging flow and reactor coolant system (RCS) sampling is used to demonstrate that the RCS is being

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borated. The licensee states that the boric acid charging flow is not a safety injection system, nor is it used for emergency boration. Therefore, we find that this variable is not applicable at the Vogtle Station.

3.3.10 Degrees of Subcooling

Within the initial submittal, the applicant has identified this as a Type A variable. As such, Table 2 of Regulatory Guide 1.97 requires Category 1 instrumentation. The applicant is supplying Category 2 instrumentation. The applicant's March 15, 1985 submittal stated that the category will be revised in Amendment 16 and that the instrumentation is in compliance with the Category 1 recommendations of Regulatory Guide 1.97.

3.3.11 Condenser Air Ejector Radiation

In the initial submittal, the applicant identified this as a Type A variable. As such, Table 2 of Regulatory Guide 1.97 requires Category 1 instrumentation. The applicant is supplying Category 2 instrumentation. The applicant states, his March 15, 1985 submittal, that the condenser air ejector radiation instrumentation is located in a non-seismically designed building and is nonseismically qualified equipment. Since it serves a backup function to the steam generator level monitors and the main steamline radiation monitor, and is not a Type A variable, the variable is revised to a Type E, Category 3. The revision and justification are acceptable.

3.3.12 Areas of Additional Noncompliance

a. Reactor Coolant System Soluble Boron Concentration.

In the applicant's initial submittal, this variable was not addressed or indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided. In the March 15, 1985 submittal, the applicant states

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that the reactor coolant system soluble boron concentration is part of the post-accident sampling system, rather than continuous monitoring instrumentation.

The applicant deviates from Regulatory Guide 1.97 with respect to this variable. This deviation goes beyond the scope of this review and is being addressed by the NRC as part of their review of NUREG-0737, Item II.B.3.

b. Radiation Level in Circulating Coolant.

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

In the March 15, 1985 submittal, the applicant states that this variable is part of the post-accident sampling system (PASS). Based on the alternate instrumentation associated with the PASS, we conclude that the instrumentation supplied for this variable is adequate and, therefore, acceptable.

c. Analysis of Primary Coolant (Gamma Spectrum)

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided. In the March 15, 1985 submittal, the applicant states that the variable is part of the post-accident sampling system.

The applicant deviates from Regulatory Guide 1.97 with respect to post-accident sampling capability. This deviation goes beyond the scope of this review and is being addressed by the NRC as part of their review of NUREG-0737, Item B.II.3.

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d. Containment Effluent Radioactivity

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

In the March 15, 1985 submittal, the applicant states that this variable is monitored by the plant vent monitor which receives discharge from the containment purge system, auxiliary building, control building and the fuel handling building. The required information is provided in the June 20, 1985 submittal and the instrumentation is acceptable.

e. Accumulator Isolation Valve Status

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

In the March 15, 1985 submittal, the applicant states that this variable is part of the ECCS valve status. The required information is provided in the June 20, 1985 submittal and the instrumentation is acceptable.

f. Reactor Coolant Pump Status

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

In the March 15, 1985 submittal, the applicant states that the reactor coolant pump status would be monitored by the reactor coolant pump circuit breaker position and the information pertaining to the instrumentation will be added to Table 7.5.2-1 in a subsequent amendment. It will be designated as a Type D

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Category 2 variable. The applicant did not provide justification for providing pump breaker position instead of the recommended motor current indication in that submittal.

The applicant provides additional information in his June 20, 1985 submittal. The justification for providing pump breaker position is that the emergency operating procedures do not utilize motor current to assess the operation of the reactor coolant pumps but pump circuit breaker positions are used. Reactor coolant pump motor current is available in the control room by means of the emergency response facility computer CRT display. This instrumentation conforms to Category 3 criteria.

Based on the current display capability in the control room, we find the deviation acceptable.

g. Pressurizer Relief Tank Level

In the applicant's initial submittal this variable was not addressed or indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

The required information for this variable was provided in the applicant's March 15, 1985 submittal and the instrumentation is acceptable.

h. Pressurizer Relief Tank Temperature

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

In the March 15, 1985 submittal, the applicant provided the required information. However, the range for the pressurizer relief tank temperature is 50° to 300°F instead of the recommended 50° to 750°F.

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The applicant stated, in his January 21, 1986 submittal, that the pressurizer relief tank (PRT) temperature range would be increased to 50 to 350°F, and that this range is adequate to monitor any expected conditions in the tank. This range relates to the tank's rupture disk that relieves pressures in excess of 91 psig. This pressure relief limits the temperature of the tank contents to saturated steam conditions under 321°F. Thus, we find this deviation from the regulatory guide acceptable.

i. High Level Radioactive Liquid Tank Level

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

The required information for the variable was provided in the applicant's June 20, 1985 submittal and the instrumentation is acceptable.

j. Radioactive Gas Holdup Tank Pressure

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

The required information for this variable was provided in the applicant's June 20, 1985 submittal. However, the applicant takes exception to the range. The range is 0 to 100 percent design pressure instead of the recommended 0 to 150 percent of design pressure. The applicant states that the design pressure of each of the waste gas decay tanks (WGDTs) is 150 psig. There are seven WGDTs per unit and two shared shutdown decay tanks. Each tank is provided with a pressure transmitter and a high pressure alarm in the control room as part of the gaseous waste processing system trouble alarm. An operator would be dispatched

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to the local control panel to clear the annunciated signal. The alarms for the WGDTs are set at 100 psig. All of the WGDTs are provided with relief valves set at or below the tanks' design pressure. The relief valves for the WGDTs discharge to shutdown decay tanks which are normally at low pressure. Should an extended discharge to the shutdown decay tank occur, a high alarm would be received prior to lifting of the shutdown decay tank relief valve. The relief valves for the shutdown decay tanks discharge to the plant vent which is monitored by the plant vent monitor. Based on the protection afforded by the installed tank relief valves and the potential eventual release being to the plant vent, the gaseous waste processing system trouble alarm and pressure indication to 150 psig in the control room is adequate to provide information concerning the status of the WGDTs.

Based on the relief valves being set below the tanks' design pressure, we find the deviation from the recommended range acceptable.

k. Emergency Ventilation Damper Position

In the applicant's initial submittal, this variable was not addressed nor indicated in Table 7.5.2-1, nor was the information required by Section 6.2 of NUREG-0737, Supplement No. 1 provided.

In the March 15, 1985 submittal, the applicant states that this variable is a part of the heating, ventilation, and air-conditioning system status. The required information is provided and the instrumentation is acceptable.

l. Auxiliary Building Radiation Level Portable Sampling

In the March 15, 1985 submittal, the applicant added the information required by NUREG-0737, Supplement No. 1, for the auxiliary building radiation level. From the information

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provided, it is assumed that this instrumentation is for the Type C variable, radiation exposure rate. The applicant deviated both in category and range. The applicant provided Category 3 instead of Category 2 instrumentation with no range specified.

Within the June 20, 1985 submittal, the applicant provided justification for this deviation. The applicant stated that the non-containment area monitors originally specified as a part of the process and effluent radiation monitoring system have been redesignated as Type E, Category 3 post-accident monitors. While the ranges of these monitors have not been expanded or increased, they provide usable surveillance data. It should be noted that Revision 3 of Regulatory Guide 1.97 (Reference 10) has reclassified this variable as Category 3. Because post-accident servicing of safety-related equipment is not expected immediately after an accident, permanently installed area monitors have not been provided. Because of the unpredictable nature of maintenance requirements post-accident, plant health physicists equipped with portable monitors will precede personnel into any areas where radiation may restrict access to service safety-related equipment. Portable radiation survey instrumentation with the capability to detect gamma radiation over the range of 10^{-1} to 10^{4} R/hr will be maintained at the health physics station.

From a radiological standpoint, if the radiation levels reach or exceed the upper limits of the range, personnel would not be permitted into the areas without portable monitoring except for life saving. Based on the alternate instrumentation used by the applicant for this variable, we find the deviations for the radiation exposure rate monitors acceptable.

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4. CONCLUSIONS

Based on our review, we find that the applicant either conforms to or is justified in deviating from the recommendations of Regulatory Guide 1.97, except for the following:

1. Containment sump water temperature - the applicant should provide fully qualified Category 2 instrumentation for this variable to provide a quantitative measure of the heat removed by way of the containment sump (Section 3.3.6).

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5. REFERENCES

1. D. G. Eisenhut, NRC letter to all Licensees of Operating Reactors, Applicants for Operating Licenses, and Holders of Construction Permits, "Supplement No. 1 to NUREG-0737--Requirements for Emergency Response Capability (Generic Letter No. 82-33)," December 17, 1982.

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13 ABSTRACT (200 words or less)

In June 1985, the staff of the Nuclear Regulatory Commission issued its Safety Evaluation Report (NUREG-1137) regarding the application of Georgia Power Company, Municipal Electric Authority of Georgia, Oglethorpe Power Corporation, and City of Dalton, Georgia, for a license to operate the Vogtle Electric Generating Plant, Units 1 and 2 (Docket Nos. 50-424 and 50-425). Supplement 1 to NUREG-1137 was issued by the staff in October 1985. The facility is located in Burke County, Georgia, approximately 26 miles south-southeast of Augusta, Georgia, and on the Savannah River.

This second supplement to NUREG-1137 provides recent information regarding resolution of some of the open and confirmatory items that remained unresolved at the time the Safety Evaluation Report was issued. This supplement also discusses some new open and confirmatory items.

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